



## Department af Reactor Technology. Annual progress report 1 Januar - 31 December 1976

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<b>Title and author(s)</b>  Department of Reactor Technology:  Annual Progress Report  1 January - 31 December 1976	<b>Date</b> June 1977  <b>Department or group</b> Department of Reactor Technology  <b>Group's own registration number(s)</b>
59 pages +      tables +      illustrations	
<b>Abstract</b>  The work of the Department of Reactor Technology within the following fields is described:  <ul style="list-style-type: none"> <li>. Reactor Engineering</li> <li>. Structural Reliability</li> <li>. System Reliability</li> <li>. Radiation Fields in Nuclear Power Plants</li> <li>. Reactor Physics</li> <li>. Fuel Management</li> <li>. Fission Product Decay Analysis</li> <li>. Steady State Thermo-Hydraulics</li> <li>. Reactor Accident Analysis for LOCA and ECC</li> <li>. Containment Analysis</li> <li>. Experimental Heat Transfer</li> <li>. Reactor Core Dynamics and Power Plant Simulators</li> <li>. Control Rod Ejection Accident Analysis</li> <li>. Economic Studies for Power Plants</li> <li>. Experimental Activation Measurements and Neutron Radiography at the DR 1 Reactor</li> </ul>	<b>Copies to</b>

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## 1. INTRODUCTION

The Department of Reactor Technology comprises six sections:

Reactor Engineering

Reactor Physics

Heat Transfer and Hydraulics

Experimental Heat Transfer

Reactor Dynamics

The DR 1 reactor

The work of these six sections during 1976 is described in this report.

During the year alternative energy became an increasingly vital topic for both public and political debate. For Risø, as an energy laboratory under the Ministry of Commerce and Energy, it seemed a natural step to increase the research efforts within development of alternative sources of energy. Towards the end of the year the Department of Reactor Technology was engaged in sun panels, soil heat storage, heat transmission through windows, windmills, district heating from power stations, and energy system analysis.

At the same time the Danish Inspectorate of Nuclear Installations was - by the end of the year - formulating a number of vital tasks to be performed by the Department. These tasks are mandatory to the introduction and acceptance of a nuclear power station. Such areas are covered as: criteria for nuclear steel pressure vessels, containment layout, and accident analysis. During the first half of the year, the Department made preparations for this type of work by studies of, e.g., the American criteria for accident analysis, and the GE response to these through GESSAR.

Furthermore, the Department took part in the Risø investigations of the consequences for Denmark (particularly Copenhagen) of a major accident at the Swedish Barsebäck reactor.

One major event of the year within reactor safety research was the US NRC-LOFT-NORHAV agreement by which the Nordic countries obtain access to American experimental results on blowdown and emergency core cooling, while the US NRC gains

access to Nordic experiments and computer models. The main effort of the Department concerning this agreement is to make best estimate emergency core cooling codes for BWRs. Seven Danes and three Nordic colleagues stationed at Risø are now developing the NORCOOL codes partly based on the earlier Danish REMI-HEATCOOL code. These codes are supplied to the US NRC and the Nordic countries within the agreement.

Finally, one reactor physicist has been stationed at the Inspectorate of Nuclear Installations, one physicist has worked on reactor dynamics at ASEA ATOM, Sweden, and one physicist on the Swedish Marviken II containment experiment on pressure oscillations.

B. Micheelsen

## **2. SECTION OF REACTOR ENGINEERING**

### **2.0. Introduction**

The main object of work in the section is to establish and maintain know-how about design, construction, operation and safety of Light Water Reactors.

This aim is pursued mainly through a strong effort on System Delineation. In addition more specific topics are dealt with, and they are described in more detail below. These are: Structural reliability, System reliability, Prediction of radiation doses to personnel in nuclear power plants, and Analysis of the failure probability of nuclear fuel elements.

General knowledge of the design and construction of reactor systems is covered under the heading System Delineation. Reports in Danish are prepared to facilitate the assessment of the merits of the different systems. General Electric BWR/6 and Kraftwerk Union PWR are investigated at present.

### **2.1. Structural Reliability**

The purpose of the work is to develop methods for evaluation of the reliability of structural components. In particular to develop computer codes, based on probabilistic methods, for evaluation of the reliability of primary components in light water reactors. The work has focused on the steel pressure vessel and the fuel element cladding (described below in 2.2).

As a supplement to a computer code for calculation of the failure probability of a steel pressure vessel (by Monte Carlo technique with Importance Sampling, PEP 706) an analytical program (ANPEP) was developed. ANPEP makes a numerical integration of the failure integral by means of discretization of all the parameters in the failure criteria. ANPEP has proved to be much faster and easier to work with than PEP 706, and it has even been able to take into account correlated variables without making excessive demands on computer memory or time.

The computer code, PFM 690, for Monte Carlo calculation of the statistical crack growth based on Paris's formula has proved the mathematical instability of this formula when the most recent experimental data on crack growth characteristics are used.

A number of international contacts and collaboration projects was established. They include manufacturers in Germany, USA, France and England.

## 2.2. Reliability of Fuel Cladding

The project is conducted in collaboration with the Metallurgy Department and it was started in June 1975. A plan over the project is shown as figure 2.1.

During 1976 the simplified fuel model "FFRS" was completed, and the performance of the model was examined by analyzing several cases, including the four EPRI Benchmark cases (described in CENPD-218) as well as experiments from the Danish irradiation programme. It was demonstrated that the model is based on sound principles, representing the state of the art in fuel modelling.

The fuel model "FFRS" is used as part of the program "FRP" to analyze the statistical distribution of fuel temperature, strain, stress etc. as function of time ( $Y_i(t)$  on fig. 2.1.). In "FRP" Monte Carlo technique or a first order Taylor approximation is utilized. Figure 2.2. shows the calculated and the measured distribution of the mid pellet strain, figure 2.3. shows the calculated contribution to the variance on the mid pellet strain for a Danish irradiation experiment.

Future work will aim at the formulation of reasonable failure criteria and collection of statistical data for the important material and design variables.

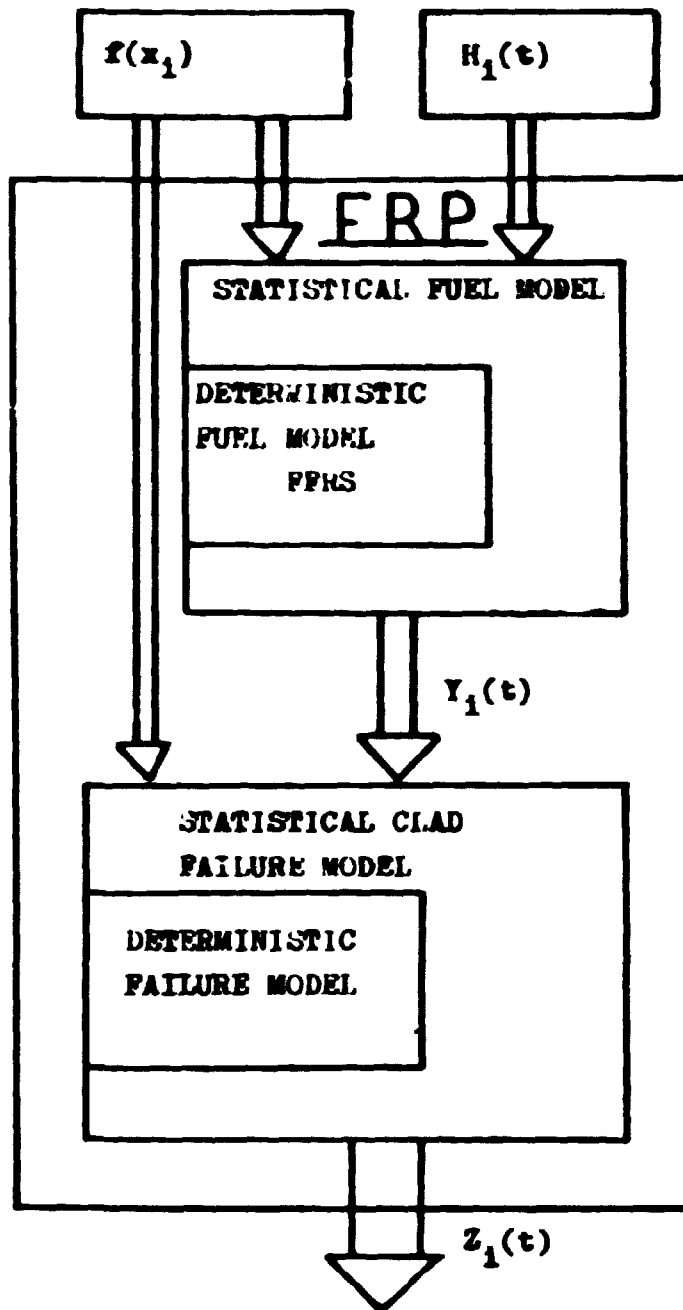
## 2.3. System Reliability

The Monte Carlo code, REDIS, developed for detailed sub-system reliability analysis has been interchanged with the Finnish program SAFTE-LR. SAFTE-LR has been tested and good agreement was found with the REDIS code. The Finish programme proved to be approximately twice as fast as ours and it can also utilize a variance reduction technique. On the other hand the Finnish programme - contrary to ours - can not be used for analysis of systems with more complicated operation like for instance standby systems.

In 1975 the Danish Automation Society formed a Working group to study the use of reliability and safety analysis



FUEL RELIABILITY PREDICTION



$x_1$ : design and material data, stochastic variables.

$H_1(t)$ : applied load on the fuel (power, flux etc.), stochastic process.

$Y_1(t)$ : clad state (stress, strain etc.), stochastic process.

$Z_1(t)$ : clad failures (stress corrosion, overstrain etc.), stochastic process.

Fig. 2.1. Diagram for fuel reliability prediction.

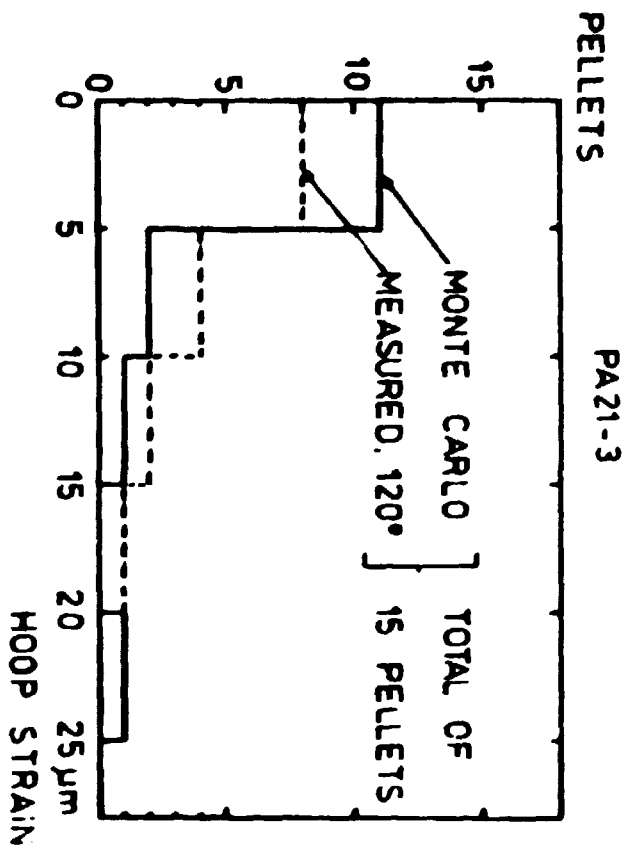


Fig. 2.2. PA21-3. Distribution of the mid pellet diameter increase.

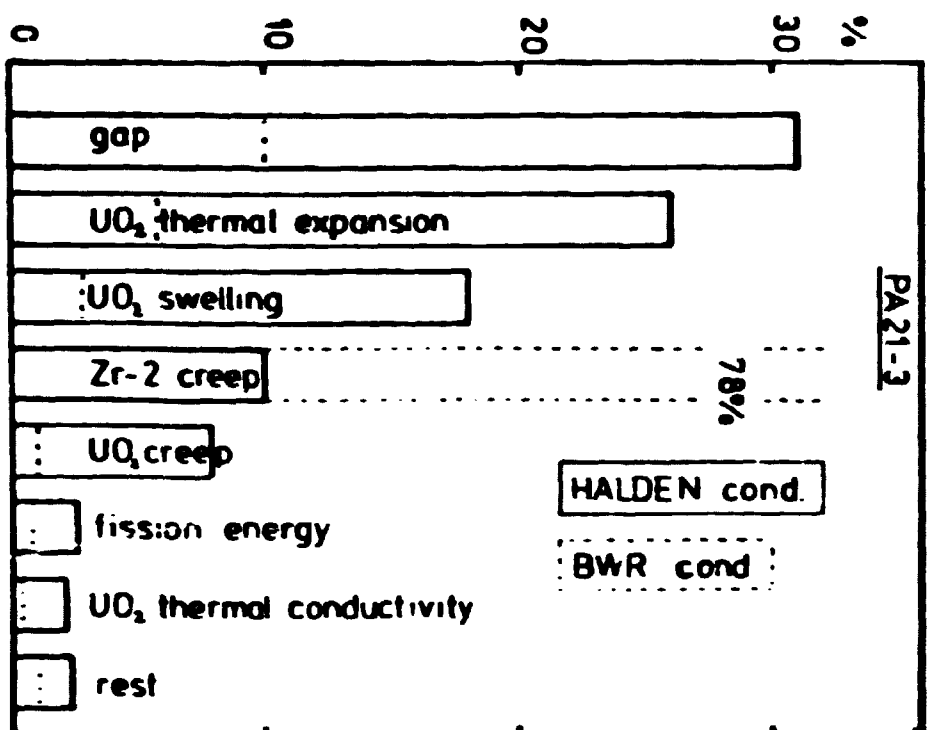


Fig. 2.1. PA21-3. Contribution to the variance on the hoop strain.

techniques in industry. One of the trial projects carried out by this group was a reliability analysis of a proposed instrument air system for a complex of fertilizer plants. The analysis was carried out before the detailed design was started and it comprised a failure analysis and a quantitative reliability analysis (Rise-M-1903).

One of the results of the analysis of the instrument air system is a series of recommendations. One of these concerns a design change of the dryer section.

The dryer section comprises two 50% units in parallel. One of these units is shown in fig. 2.4 . It consists of two adsorbers, 1 and 2. A timer positions the pilot operated valves in the inter connecting pipe system. These valves are operated in such a way that one dryer is always drying, the other being regenerated.

Every fourth hour the regenerated dryer is switched over to drying operation and vice versa. The failure analysis has proved that with a frequency of approximately 2/year, one of the valves SV1-4 will fail to close when required, so that the inlet air will flow directly to the atmosphere. This will cause a failure of the instrument air supply and a shut down of the entire fertilizer plant.

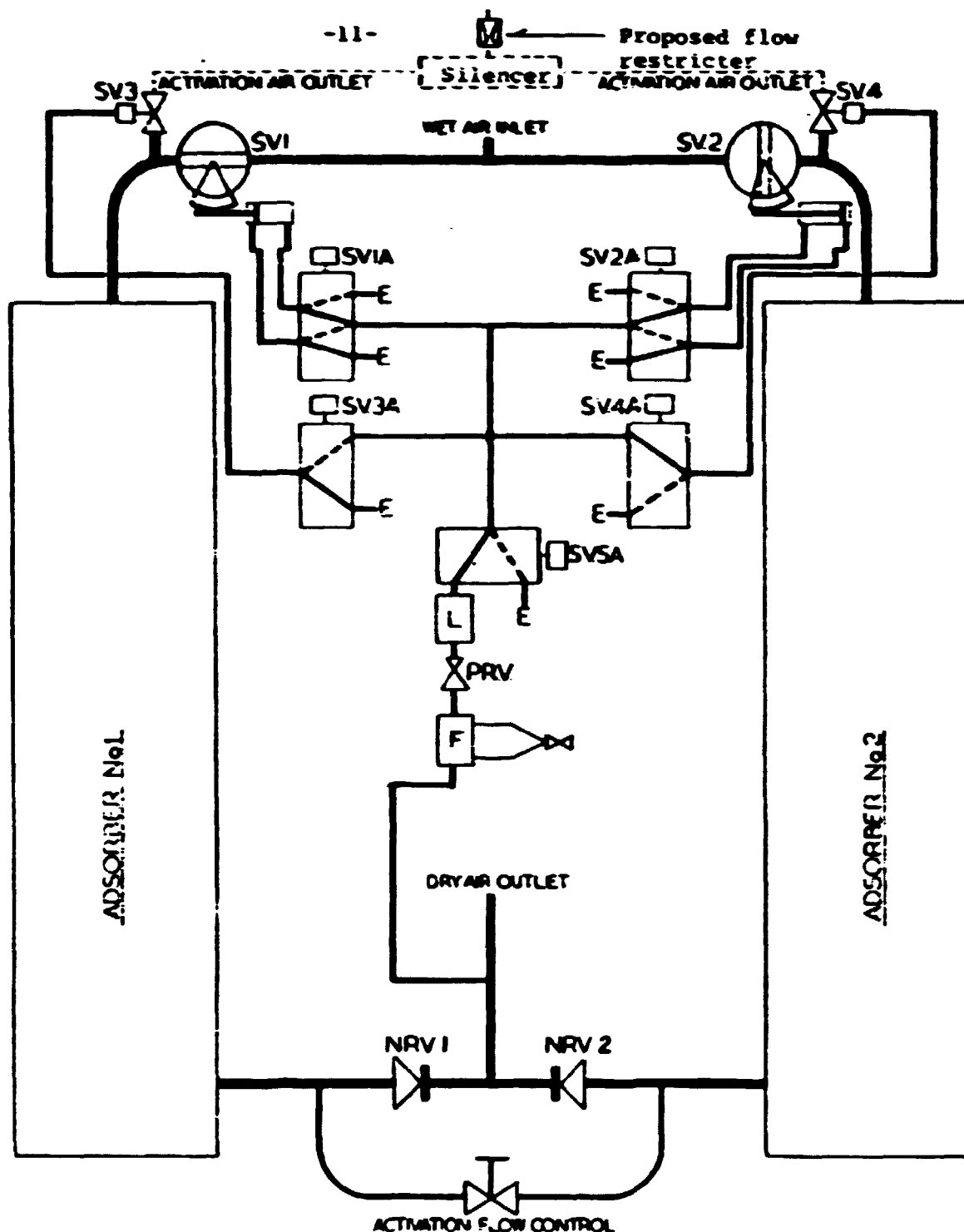
These serious consequences will be avoided by the proposed change of the dryer design: insertion of a venturi nozzle in the activation air outlet pipe, (see fig.2.4.). Such a device will act like the flow restricters in the main steam lines of a BWR: The normal pressure drop across the nozzle is negligible and during the above valve failure the critical flow through the nozzle will be less than the capacity of a standby compressor and will not influence the instrument air supply.

The REDIS program has been tested by a reliability analysis of the most complex part of the system: the compressor system. The program proved to be easy to fit to the special operating conditions for this system, (SRE-4-77).

#### 2.4. Assessment of Radiation Fields in Nuclear Power Plants

A model for the build up of radiation fields around power plant components during normal operation has been completed. The type of power plant chosen for the study is one with a General Electric BWR/6.

---



- |             |  |
|-------------|--|
| SV1 & SV2   | BUTTERFLY TYPE AIR OPERATED, PILOT CONTROLLED, ADSORPTION VALVES |
| SV3 & SV4   | PISTON TYPE AIR OPERATED, PILOT CONTROLLED, ACTIVATION VALVES    |
| SV1A & SV2A | 4-WAY SOLENOID OPERATED, TIMER CONTROLLED, PILOT VALVES          |
| SV3A & SV4A | 3-WAY SOLENOID OPERATED, TIMER CONTROLLED, PILOT VALVES          |
| SV5A        | 3-WAY SOLENOID OPERATED, 'BREAKDOWN' VALVE                       |
| PRV         | PRESSURE REDUCING VALVE (IF REQUIRED)                            |
| F           | AIR FILTER (IF REQUIRED)   |
| L           | OIL FOG LUBRICATOR   |

Fig. 2.4. Dryer flow diagram.  
(from PISZ-M-1903)

In the model the reactor and turbine systems are represented by 39 components as shown in figure 2.5. The inventories of radioactive fission and corrosion products in these components are described by a system of first order linear differential equations of the form

$$\frac{dn_{ij}(t)}{dt} = P_{ij}(t) - R_{ij} n_{ij}(t), \quad (1)$$

where  $n_{ij}(t)$  is the number of atoms of nuclide  $j$  in component  $i$  as a function of time,  $t$ .  $P_{ij}(t)$  is a "production term" describing the creation of  $j$ -atoms in the coolant of component  $i$  by the following processes: release of fission products from fuel, release of corrosion products from construction material, mass transport from adjacent components, release of material deposited on component walls, neutron activation, and decay of radioactive precursors. The term  $R_{ij} n_{ij}(t)$  describes the removal of  $j$ -atoms by the processes: mass transport to other components, deposition on walls, neutron absorption, and radioactive decay. For fission products one decay chain is considered at a time. As a typical decay chain consists of 4-5 significant nuclides, the system of equations (1) will often have the size of  $39 \times 5 = 195$  equations. For corrosion products the system of equations always consists of  $39 \times 2 = 78$  equations, because only two nuclides are considered simultaneously, namely a non-active target nuclide and its radioactive activation product nuclide. The integration of equations (1) is performed numerically by the FORTRAN code FICOPI (Fission and Corrosion Product Iinventories). Another code, INAPI (Intrinsic Activation Product Iinventories), calculates the contents of nuclides formed by activation of the coolant itself. By adding together the contributions from all significant fission products, corrosion products, and intrinsic activation products, the total activity inventory of the components is obtained. The radiation fields outside the components are calculated by means of a relatively simple shielding code, SHIELD, based on point-kernel technique. The results obtained with the model so far show reasonable agreement with measurements reported from operating power plants. A detailed description of the model is given in Risø Report No. 353.

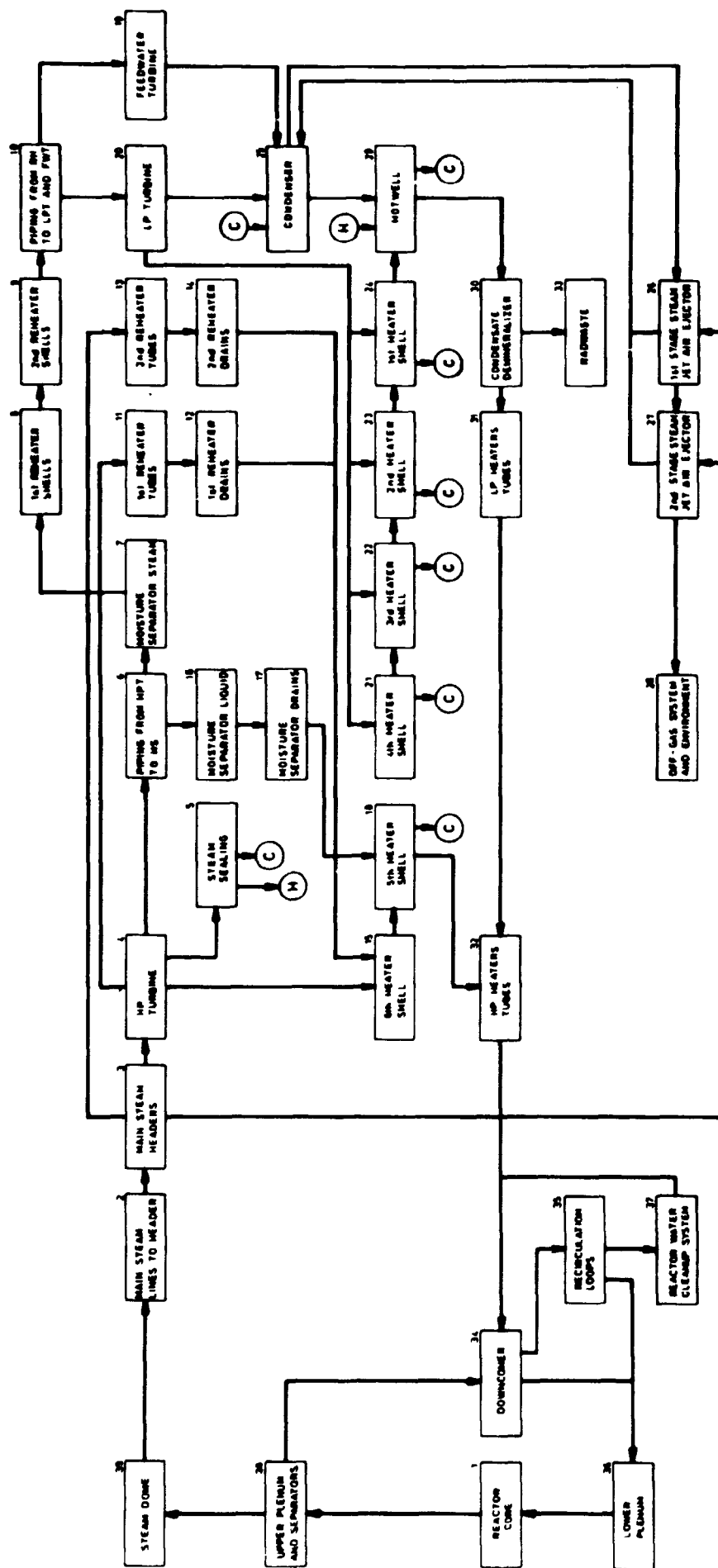


Fig. 2.5 The power plant model.

### 3. SECTION OF REACTOR PHYSICS

#### 3.0. Introduction

The reactor physics is dealing with the behaviour of the neutrons in the reactor and the main results from the analysis are the reactivity, the power distribution in the reactor core, and the burn-up of the fuel.

In the period reported here development work has been concentrated on the following topics:

1. Management and economy of the fuel cycle.
2. Fission product and actinide inventory in power reactors with regard to both health hazards and decay heat in case of reactor accidents.

An important part of the work going on in the section is the continuous testing of the reactor physics programme system for light water reactors, which is partly carried out in collaboration with a reactor vendor; but the testing has also included international benchmark calculations. The computer programme system is in a period of consolidation and is not reported on in the following.

#### 3.1. The Fuel Management Program SOFIE

The program SOFIE is a very flexible tool for investigation of the various aspects of the fuel management problem. The fundamental features of this multicycle fuel management program is a simplified reactor physics treatment of the burn-up combined with a rather extensive treatment of the economic aspects of the fuel cycle. The reactor physics and the economic parameters are brought on a form suitable for optimization by linear programming technique. Apart from the complete fuel cycle optimization the program can be used for economic calculations only, or it can be used as a burn-up program coupled with an economic analysis but leaving out the optimization.

As mentioned above the reactor physics model is rather simple. The core is divided into radial regions, in which the average power densities and minimum end-of-cycle reactivities for each region and cycle are assumed known. Fuel elements with the same operating history are grouped in mocs (mutual operating

conditions). A moc can have the following history: A number of fuel elements are loaded into the outer region of the core at the beginning of the second cycle, and at the beginning of the third cycle they are shuffled to the inner region of the core, where they remain until they are discharged at the end of the fourth cycle and later reprocessed. The user of the program must specify a reasonable number of mocs in order to get a reasonable answer from the program. On the basis of the location in the different regions, the burn-up of the mocs is calculated using a simple correlation for the power sharing among the mocs in the same region. When the burn-up of the mocs is known, the price of one element of each moc can be calculated. The program requires that lead and delay times, and the time variation of the price, are specified for all components of the fuel cycle. The prices are transferred to the same point in time using the present worth method.

The number of elements in each moc is determined by use of linear programming technique. The present worth of the fuel cycle cost over all cycles is used as object function, which is minimized subject to linear equality or inequality constraints.

The object function and the constraints have the following form.

$$\text{object: } \min \sum_{i=1}^N C_i X_i \quad \text{subject to the following constraints}$$

$$\sum_{i=1}^N X_i \delta_{i,j,k} = N_k$$

$$\text{and } \frac{1}{N_k} \sum_{i=1}^N X_i k_{\infty,i,j} \delta_{i,j,k} \geq k_{\infty,eoc,j,k}$$

where  $N$  denotes the total number of mocs

$C_i$  the present worth value of one fuel element from moc  $i$

$X_i$  is the number of elements in moc  $i$

$$\delta_{i,j,k} = \begin{cases} 1 & \text{if moc } i \text{ is defined in cycle } j \text{ region } k \\ 0 & \text{for all other} \end{cases}$$

$N_k$  is the number of elements in region  $k$



$k_{\infty,i,j}$  is the end-of-cycle reactivity of elements in moc i at cycle j  
 and  $k_{\infty,eoc,j,k}$  is the minimum end-of-cycle reactivity for region k at cycle j

The constraints are generated for each cycle and region. Other constraints may be applied such as maximum allowable beginning-of-cycle reactivity, the total reactivity of the core, and constraints which will give an equilibrium fuel cycle after a certain cycle.

The burn-up of the fuel elements depends on the actual mix of elements. After the number of elements in each moc has been determined in the linear programming section the burn-up and cost data for the mocs belonging to the solution are recalculated. The resulting change in the object function and the reactivity constraints, may require a new solution to the linear programming problem. Experience has shown that in most cases two iterations are sufficient to obtain a stable solution.

The solution may be utilized in various ways. Physical characteristics such as actual loading scheme for the core and reactivity at beginning and end of cycle may be written out. Table 3.1 shows the top of such a table generated by SOPIE. The solution may also be used for generating cash flow tables for all components of the fuel cycle. For the time span investigated the following tables are written out (if wanted): yellow cake procurement, conversion, enrichment, fabrication, transport of fresh fuel, transport of irradiated fuel, reprocessing, reconversion and reprocessed plutonium. The numbers in the tables are given for each year. The tables are collapsed into three main tables with the following headings. "Total cost fresh fuel", "Total cost irradiated fuel" and "Total cost". Figur 3.1 shows a graphical representation of the last mentioned table. An important quantity produced by this kind of programs is the cost of generating one kWh. The SOPIE program calculates this quantity for each cycle and for the total time span considered (Figure 3.2). The time averaged kWh-price is split into the components that make up the price, as shown in table 3.2.

Table 3.1

Number of elements in each batch and burn-up of batches

Batch	Number of elements	Burn-up GWD/TU	Incremental burn-up in the cycles				
			cycle 1	2	3	4	5
1	154	8.8	8.8				
2	195	17.3	8.7	8.6			
3	167	21.5	6.3	7.2	7.9		
4	76	20.0	4.8	4.8	5.1	5.2	
5	81	26.8		9.4	9.1	8.3	
6	73	28.6		6.6	6.7	8.7	6.6
7	88	26.8			9.5	9.0	8.4
8	107	28.1			6.6	6.6	7.4
9	55	26.8				9.4	9.1
10	112	28.0				6.8	6.7

Table 3.2

Time average kWh-price

Price 1 kwh	kr/kwh	%
uranium, fresh ( $U_3O_8$ )	0.010908	46.38
conversion ( $U_3O_8 - UF_6$ )	0.000504	2.14
separative work, fresh	0.007376	31.36
fabrication	0.003286	13.97
transport of fresh fuel	0.000000	0.00
transport of irradi. fuel	0.000000	0.00
reprocessing	0.004597	19.54
reconversion	0.000000	0.00
uranium, reprocessed	-0.003284	-13.96
separative work, reprocessed	-0.000870	-3.70
plutonium, reprocessed	-0.002110	-8.97
interest out of core before use	0.000963	4.10
interest in core	0.002071	8.80
interest out of core after use	0.000079	0.34
total	0.023522	100.00
1 Dkr. = 100 øre = 0.167 US \$		

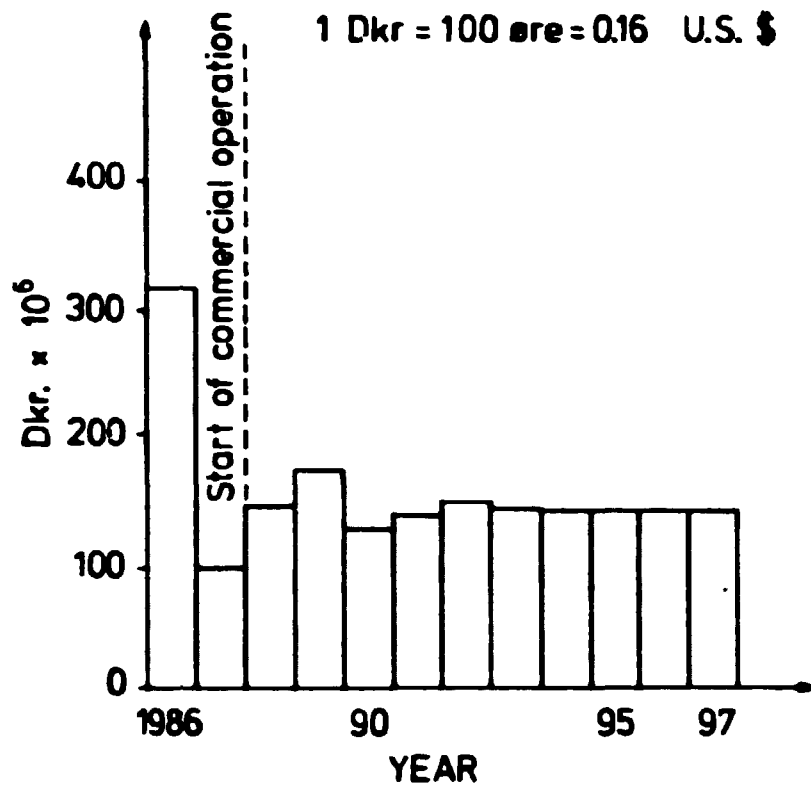


Fig.3.1 Cash flow for the fuel cycle

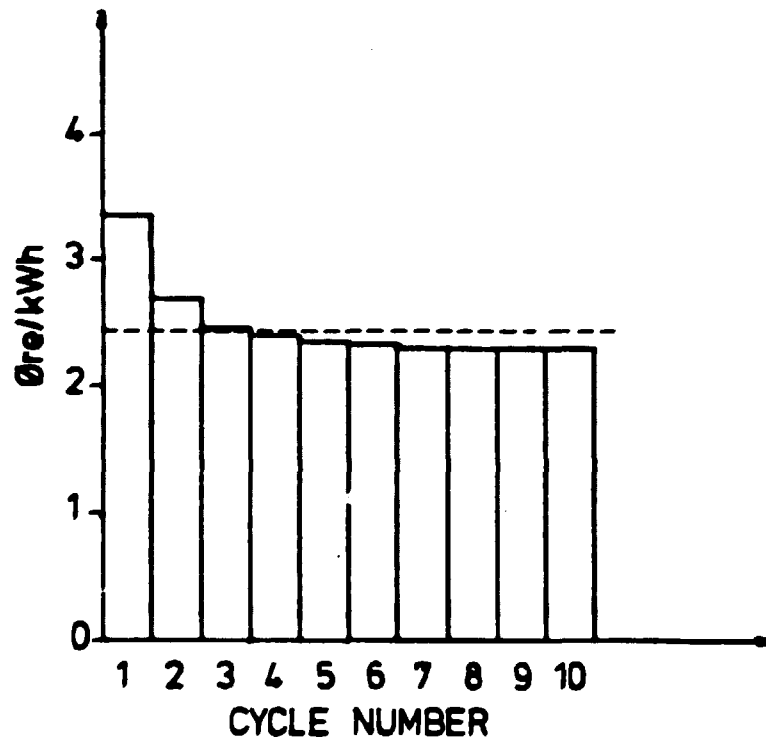


Fig.3.2 Øre/kWh for the single cycle

### 3.2. Fission Product and Actinide Density Calculation

In order to evaluate the health hazard from fission products, actinides and their daughters in case of a reactor accident a program has been developed that calculates the concentrations of a large number of nuclides during power production in a light water reactor fuel box. The program may also be used to follow the transformation of high-active waste on a large time scale.

The usual set of differential equations for the concentrations of the different nuclides is easily solved numerically under the simplifying assumptions that both the flux and the precursor concentration remain constant during the timestep. The latter assumption is justified, when the larger time interval of assumed constant flux is subdivided into timesteps starting with a short one - of order of the shortest half life occurring - and doubling the step length at each subsequent time step.

The fission product calculation comprises 380 nuclides under the following transformations: direct yield from fission, neutron capture and internal transitions.

In the calculation of the actinide densities 140 different nuclides with atom numbers between 80 and 100 are treated, subject to the following transformations: neutron capture, fission, beta-decay, alpha-decay, internal transition, electron capture, (n,2n) processes and spontaneous fission.

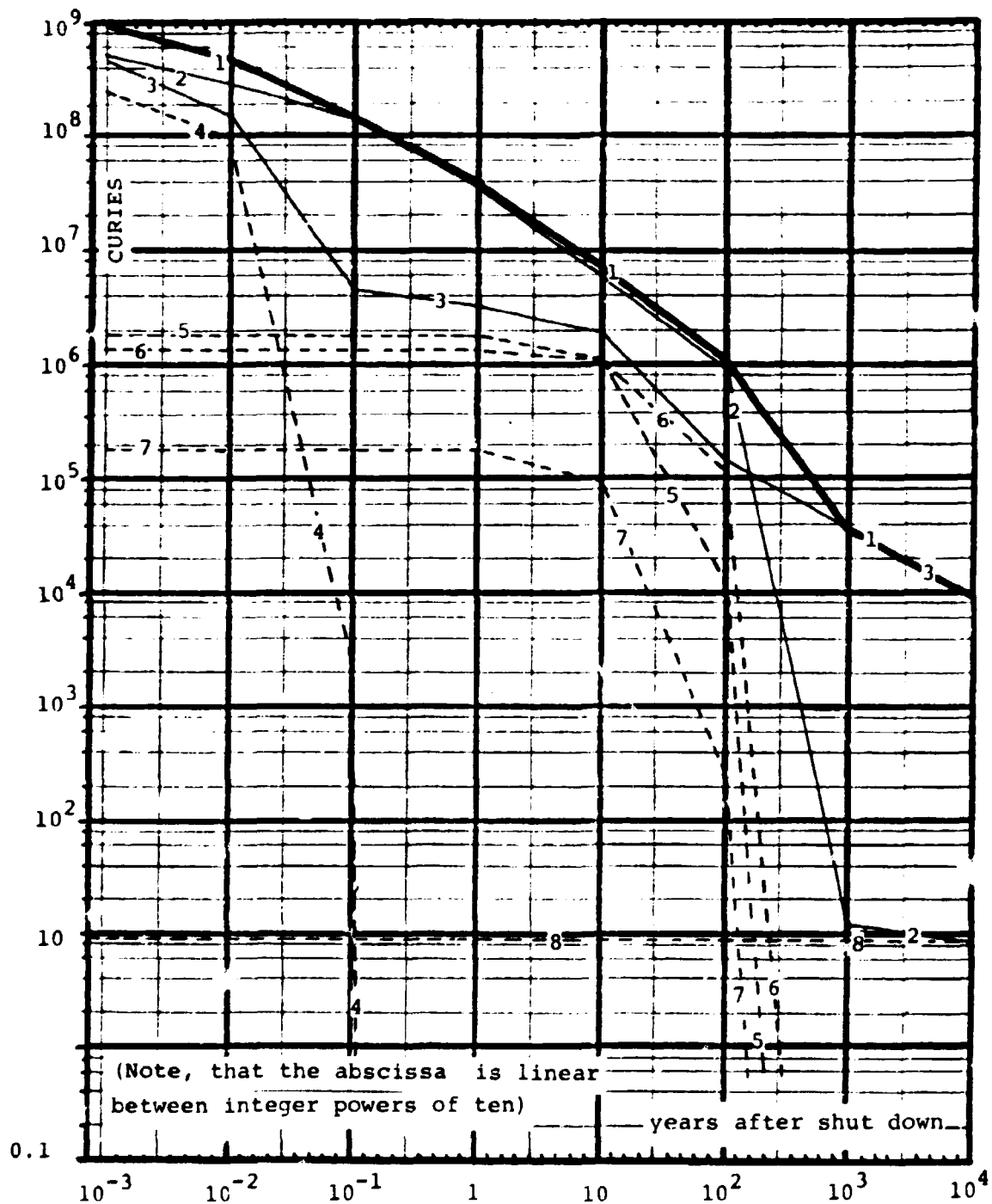
As an illustration the decay of fission products and actinides with daughters after shut down of a reactor is shown in Fig. 3.3. The time axis is graded linearly between integer powers of ten. Thus in any interval ( $10^n - 10^{n+1}$ ) the scale is semi-logarithmic, and consequently exponential decays are depicted as straight lines. This means that the straight lines in Fig. 3.3 are good approximations in decades where one decay constant is dominant.

### 3.3. Decay Heat Calculations

563 different nuclides, including metastable states, are treated in the decay heat calculations. Decay schemes are known for 287 of these nuclides.

The nuclides with unknown decay schemes are all far from the line of beta stability, and beta decay can in this

Radiation intensity of the spent fuel from  
the production of 1000 eMW years in a LWR.



1. Total activity of the fuel.
2. Activity of all fission products.
3. Activity of all actinides (and their daughters).
4. Activity of  $\text{Np}^{239}$  (Half life= 2.35 days).
5. Activity of  $\text{Pu}^{241}$  (Half life=14.0 years).
6. Activity of  $\text{Sr}^{90}$  (Half life=29.0 years).
7. Activity of  $\text{Kr}^{85}$  (Half life=10.7 years).
8. Activity of  $\text{Cs}^{135}$  (Half life= $2.3 \cdot 10^6$  years).

case proceed to a large number of excited levels in the daughter nucleus. The description of the beta decay is therefore based on a concept known from nuclear reaction theory, that of a strength function.

The strength function used here is assumed equal to zero below a cut-off energy and equal to a constant above the cut off energy. The cut-off energy is equal to  $0.13/A$  MeV and  $26/A$  MeV, respectively, for even-even, odd mass and odd-odd nuclides.  $A$  is the atomic weight.

The probability of beta decay to an excitation level is calculated using the above described method. It is assumed that decay from the excited level to the ground state takes place by emission of just one gamma quantum. In this way a decay scheme for each nuclide is established.

Calculations of decay heat has been compared with experiments. The results are shown on fig. 3.4.

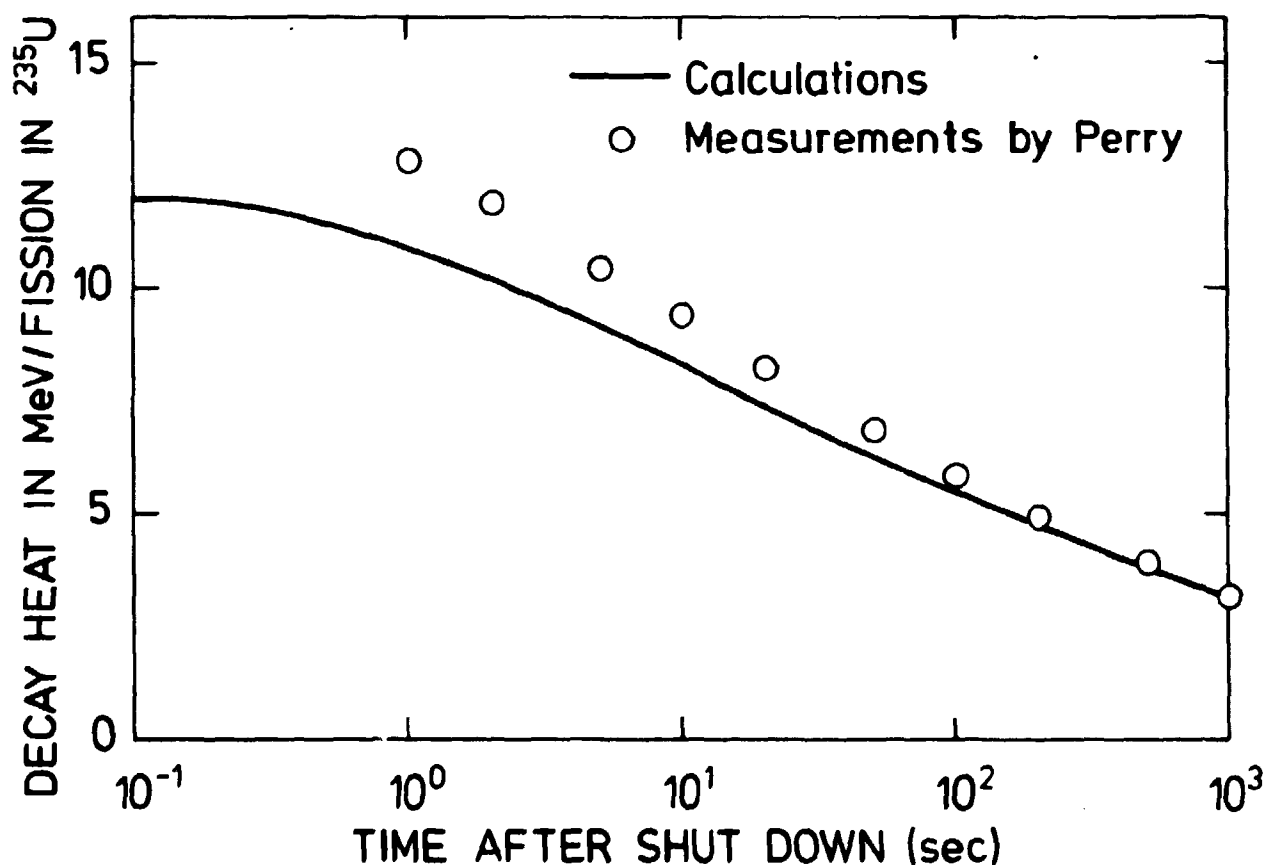


Fig. 3.4 Decay vs. time after shut down of a reactor

#### 4. SECTION OF HEAT TRANSFER AND HYDRAULICS

##### 4.0 Introduction

The main efforts of the section are directed towards the development of computer models for thermodynamic and hydraulic phenomena in nuclear power reactors. Some efforts are, however, spent on a more general application of the section's knowledge in the fields of heat transfer and fluid dynamics.

The main working areas are:

1. Steady state reactor thermo-hydraulics.
2. Reactor accident analysis.
3. Participation in international safety related experiments.
4. Low temperature heat storage, low temperature solar energy and related topics.

##### 4.1. Steady State Reactor Thermo-hydraulics

The development of a film flow model for the prediction of burn-out in tubular and annular geometries has been continued this year.

The new "microscopic" model which was mentioned in the last annual progress report, has been improved in several ways. The net deposition rate of droplets is now described by the difference between the actual mean droplet concentration and the mean droplet concentration in the equilibrium state. The slow radial diffusion of droplets has been taken into account by dividing the gap in the annular geometry into two subchannels. These improvements give, as shown in SHH-12-76, good predictions of the data set available for an adiabatic developing film flow in annular geometries. The model also gives a reasonable interpretation of data, where the tube film carries much more liquid per unit perimeter than the rod film. These data were previously supposed to be equilibrium data and the deposition process was assumed to be responsible for the asymmetry. The prediction made by the improved model, however, indicated that the data were in fact non-equilibrium data. The asymmetry could, therefore, be explained by asymmetric inlet conditions. An experimental verification of this hypothesis is supposed to be achieved, when the experimental data from the high pressure water loop have been analysed.

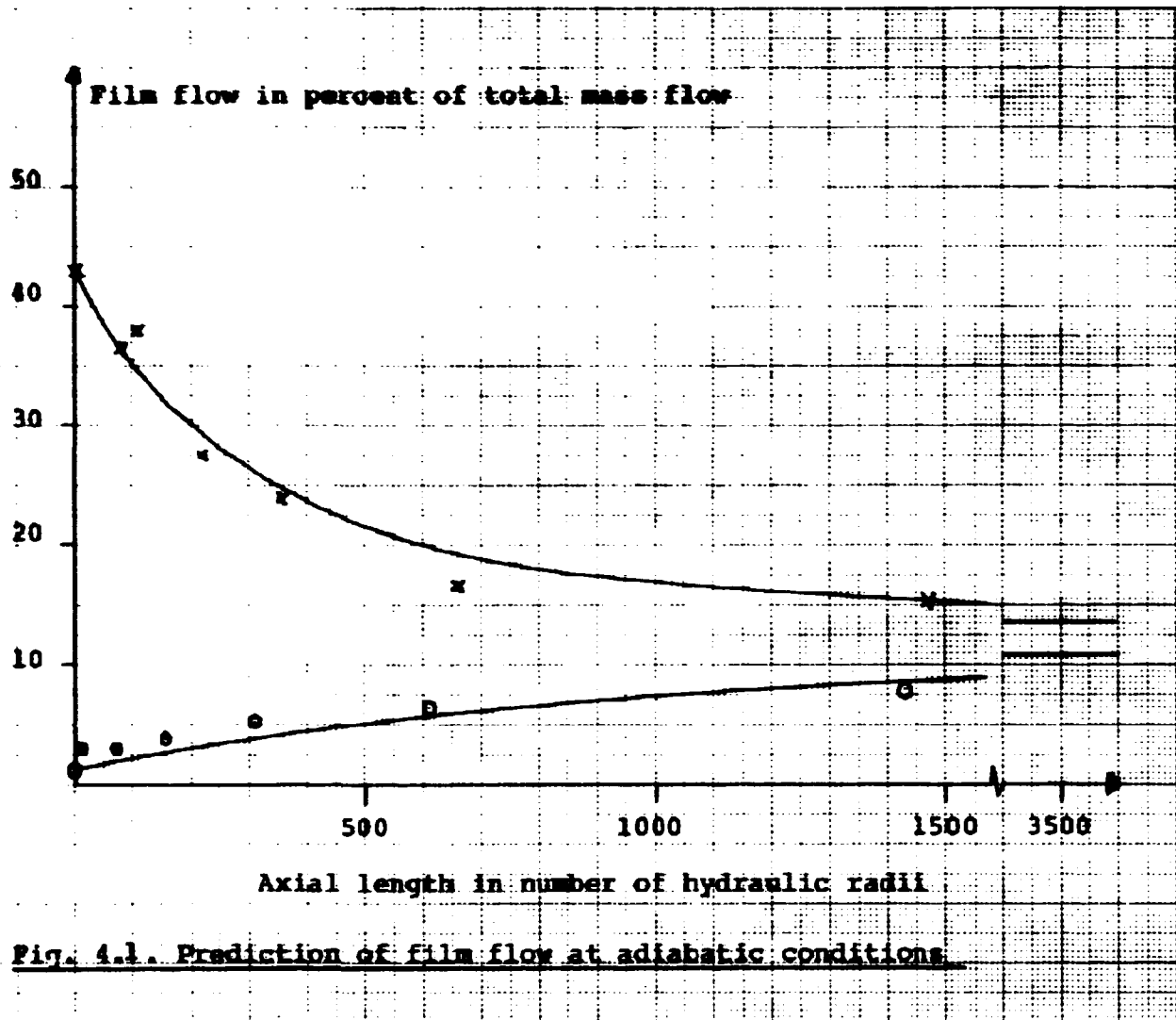


Fig. 4.1. Prediction of film flow at adiabatic conditions.

x Experimental tube film flow from AECL-3656  
 o Experimental rod film flow from AECL-3656  
 — Predictions by the film flow model

Length of the annular test section	2.90 m
Radius of the rod	9.9 mm
Radius of the tube	11.9 mm
Hydraulic radius	2.0 mm
Pressure	35 bar
Mass flux	1360 kg/m <sup>2</sup> /s
Steam quality	32 %

The prediction shows, that the state of equilibrium is reached at approximately 3500 hydraulic radii i.e. 7 m.



## 4.2 Reactor Accident Analysis

The work has included both development of computer codes for LOCA analysis and basic studies in support of the code development. Most of this work has been done within the framework of the NORHAV project.

Furthermore, work in connection with safety analysis of existing reactors has been undertaken.

### A. Basic Studies

The properties of the basic equations for two-phase flow have been studied both theoretically and in practice - by integrating various engineering forms of the conservation equations.

The TPD (Two-Phase-Dynamics) code is based on a drift-flux model which permits thermodynamic non-equilibrium for the water phase.

The code has been used as a research tool in support of the development of TINA, the dynamic subchannel code for blowdown calculations. TPD has also been used for calculations of critical flow.

The basic properties of two-fluid models (six conservation equations) have been studied theoretically with particular emphasis on the influence of the "virtual mass" effect. It was found that the "continuity waves" can be stabilized - even in the absence of dissipative terms.

The properties of the Turner and donor-cell numerical schemes for integration of the conservation equations were investigated both theoretically and in practice. The RISQUE-T computer code - a two-fluid model employing the Turner scheme - was developed and tested - thus proving the feasibility of using this model for practical calculations. RISQUE-D a two-fluid donor-cell code is presently being coded.

### B. Blowdown

The TINA code performs dynamic subchannel calculations for the blowdown phase of a LOCA. The two-phase flow is described by a drift-flux model which permits thermodynamic non-equilibrium of the water. The numerical method is implicit and permits the use

of large time steps without problems of stability. The program was transferred to the CDC computer at ANC, Idaho, and extensively tested on the Semiscale PWR/blowdown experiments.

The results were very encouraging although a complete verification was not possible due to the rather poor definition of the experimental boundary conditions for the Semiscale core.

### C. Core Heat-up

CORECOOL, which is a further development of REMI/HEATCOOL, is a computer code for analysis of spray cooling transients in boiling water reactors. CORECOOL is developed in co-operation with the General Electric Company.

During the first half of the year a physicist from the section was working at General Electric. During this period the development of the code was finished, and a first testing and sensitivity analysis of the code was carried out. CORECOOL is described in NEDO-21325, which was presented at the CSNI meeting in Spätind, Norway, in September.

Fig. 4.2 shows a comparison of the peak cladding temperature (PCT) vs. time, between test results and CORECOOL calculations with same initial conditions.

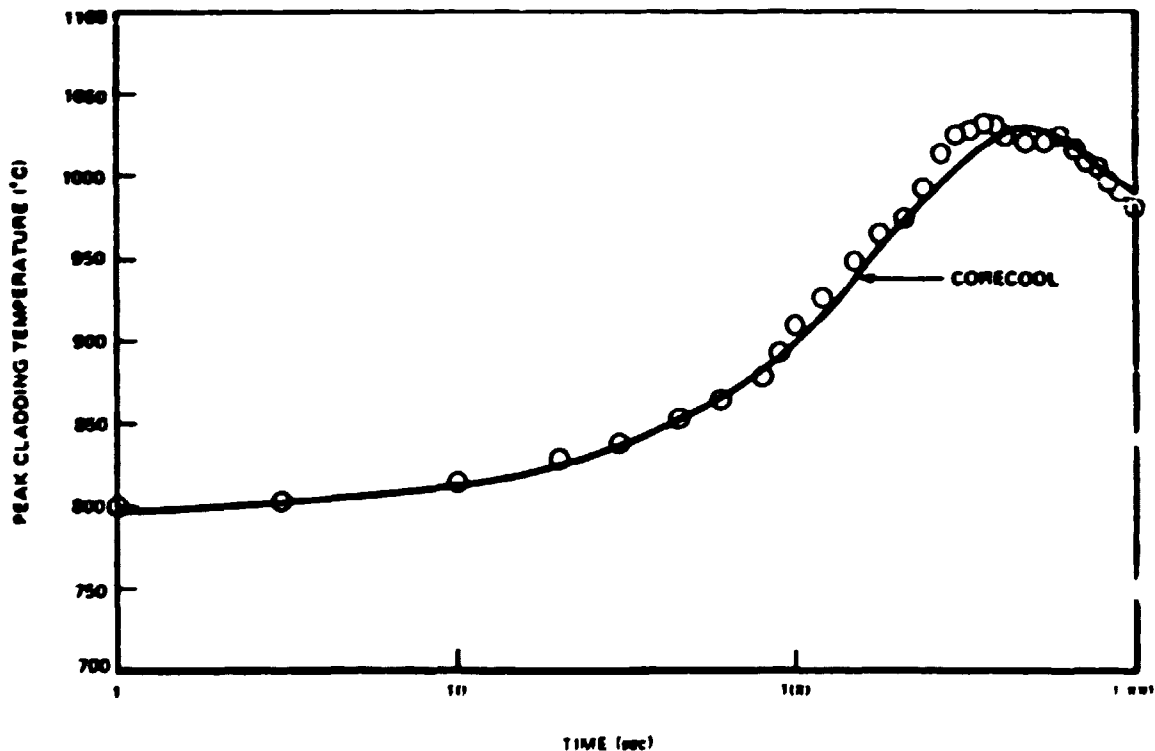


Figure 4-2. Peak Cladding Temperature for BWR Fuel Element

In the fall the existing Nordic co-operation NORHAV was extended to include the US NRC. Through this co-operation the NORDIC countries obtain the results of the LOFT and Semiscale experiments. On the other hand the NORHAV-project will furnish the NRC with advanced best estimate computer models for LOCA-analysis of BWR. This project covers the development of the NORCOOL computer code. In connection with this co-operation three physicists from Finland, Norway and Sweden are stationed at Risø.

As a result of discussions with the NRC it was decided to divide the code development into two phases: The first phase aims at the development of a simple reflooding and spray cooling model, NORCOOL-I, based on an earlier development, the REMI/HEATCOOL code. This phase will be finished in the beginning of 1977. The second phase in the NORHAV-NRC co-operation consists of the development of a more sophisticated and detailed best estimate model including parallel channels and multi-dimensional flow. This phase, NORCOOL-II, will be finished in 1978.

The theoretical basis for NORCOOL-I was developed in the third quarter of the year, and it is described in NORHAV-D-29, which was presented at the fourth Water Reactor Safety information meeting in Washington in September. In the fourth quarter the computer code was designed and the mathematical models and numerical methods developed. This includes models for the two-phase flow and heat transfer below the two-phase levels, the movement of the levels, and comprehensive changes in the existing code. The coding of the code was finished in December, and the testing will take place in the first quarter of 1977.

In connection with the NORCOOL-project a new model for thermal radiation has been developed. The model, which is semi-empirical, includes absorption and emission from the two-phase mixture.

#### D. Analysis of Containment Response

Since spring 1975 a post-graduate study has dealt with the development of a computer code for the simulation of the containment response to Loss-of-Coolant-Accidents.

The model is a multiroom model, where rooms and room connections are represented by one-dimensional pipes, connected in a suitable manner into a pipe network.

The six basic differential equations describing the two-component, two-phase fluid dynamics are:

1. Three mass conservation equations for air, water and steam.
2. One momentum conservation equation for the mixture of the three above mentioned components.
3. Two energy conservation equations for air, steam (the gas mixture) and water.

Furthermore, the model is based upon the following description of the physical processes:

- a. Air is an ideal gas, which obeys the ideal gas equation. Together with steam it forms a gas mixture of independent constituents, which follows the Dalton rule.
- b. Steam may be saturated or superheated.
- c. Water may be subcooled, saturated or superheated.
- d. The rate of evaporation depends upon the water superheat, described by a "flashing" correlation, the rate of condensation has not yet been built in.
- e. The slip is accounted for by the drift-flux model.

The system of differential equations is solved by a modified form of the linearized, implicit method by Turner.

The code is now in a testing phase. The code verification will soon start, and experimental data from among others, Marviken I, Marviken-II-CRT and TECPO, are expected to be the base for comparison.

#### E. Barsebäck Investigations

A study of the possible consequences for Danish territory of hypothetical severe accidents with core melt-down in the Swedish Barsebäck BWR has been undertaken at Risø. The results have been published in the report Risø-M-1905. The study was partly based on WASH-1400. The Department of Reactor Technology participated with an evaluation of the Barsebäck core inventory and a study of the influence on release fractions of the differences in design between the Peach Bottom Reactor (used in WASH-1400) and the Barsebäck Reactor.

#### 4.3 Participation in Experiments Abroad

A. The Marviken Containment Response Tests, MXII-CRT, is the second project in the international cooperation on full-scale containment experiments conducted in the pressure suppression (PS) containment of the abandoned nuclear power station at Marviken, Sweden. The project is made in collaboration between the German Federal Republic, Holland, the United States of America, Japan, and the four Scandinavian countries, while France withdrew at an early stage, as it became clear, that they had given up the BWR-line.

The primary objective of the project is to provide experimental results on containment pressure oscillations, like those previously observed in the blowdown experiments of Marviken I, in order to improve the understanding of these oscillations.

The test program including nine blowdowns (one more than originally planned) was performed during the period from February to October. The final reporting, which is scheduled to six months, will thus be completed in the spring of 1977.

B. TECPO (Theoretical Efforts on Containment Pressure Oscillations) is a Nordic project associated with the MX-II-CRT project described above. It is intended via joint theoretical and experimental efforts to obtain a deeper insight into the mechanisms governing the pressure oscillations.

The experimental investigations are carried out in a small scale model of a PS-containment. The first series of sixteen blowdown tests was carried out before the start of the MXII experiments. The primary purpose of these tests was to investigate the influence on the pressure oscillations of various parameters such as flow rate and composition, geometrical configuration, pool temperature, etc. The second series of tests comprising eight blowdowns was started after the termination of the MXII experimental phase. The purpose of these tests is to investigate the efficiency of various means suggested for the mitigating of pressure oscillations.

The theoretical efforts have been limited, because the experiments were more time-consuming than expected.

It has been possible to predict eigenfrequencies of the scale model (as well as the lower eigenfrequencies of the large scale facility) by regarding the containment as a system of coupled pipes with standing (acoustic) waves. The wave forms thus calculated compare favourably with the corresponding wave forms based on spectral analysis of the measured pressures. However, it has not yet been possible to develop a satisfactory condensation model, which is necessary for the prediction of pressure amplitudes.

#### C. ISPRA Blowdown Project

A PWR blowdown loop (LOBI) is presently being constructed at Ispra. The German government finances the construction of the loop and part of the running expenses during the experimental phase.

Two meetings (number 7 and 8) of the "Ad hoc Specialist Working Group on Part B of the ISPRA Blowdown Project" were held during the year. The section was represented at meeting no. 8.

#### 4.4. Low Temperature Heat Storage, Low Temperature Solar Energy and Related Topics

In connection with the growing interest for alternative energy sources some work has been done on low temperature heat applications. A finite element computer programme is being modified in order to study the possibilities of storing low temperature heat in a soil reservoir. A computer model of a solar heating system including sunpanels, buffer tank, heat consumption and supplementary heat sources is under development.

A study of combined electricity and district heating production has been started supplemented by an investigation of district heating in connection with the burning of waste as well as low temperature heat storage concepts.

The section has at the request of the Chemistry Department made some preliminary examinations of the temperature distribution around a cylindrical cavity containing high level solid radioactive waste. The waste was assumed to be buried in great depth in clay.

The calculations were carried out after different periods of decay before disposal.

The calculations confirmed that the waste should decay some years before disposal. Further examinations are necessary before conclusions can be drawn.

## 5. SECTION OF EXPERIMENTAL HEAT TRANSFER (SEHT).

### 5.0 Introduction

The section performs experimental research on heat transfer problems in nuclear power plants. The work is done in close co-operation with the theoretical work of the Section of Heat Transfer and Hydraulics.

### 5.1 High Pressure Water Loop

A new more efficient safety valve has been installed in the loop.

For use at the film flow measurements an ejector pump has been constructed and placed in the by-pass at the main pump.

Due to severe problems with electric noise from the two 250 kW thyristor regulated power supply units each of the units was equipped with an inductance coil of 150  $\mu$ H. These coils choked the very high peaks from the thyristors and solved the problem of electric noise in the measuring channels.

A true rms device to measure the odd shaped current and voltage in the power supply units has been constructed and tested against two very expensive commercial equipments. The deviations were within the accuracy of the instruments.

After a recalibration of the thermocouples and the flow measurement system a series of heat balance measurements was carried out in which the thermal power was compared with the electric power. The deviations were within a few percent.

Regular test runs were started late in the year using a test section of annular cross section with both sides of the annulus heated.

### 5.2 Film Flow and Pressure Drop Measurements

The purpose of the measurements is to provide data for the development of the film flow model to prediction of burnout in tubular and annular geometries (cf. 4.1).

Fig. 5.1 Mounting of test section ➡





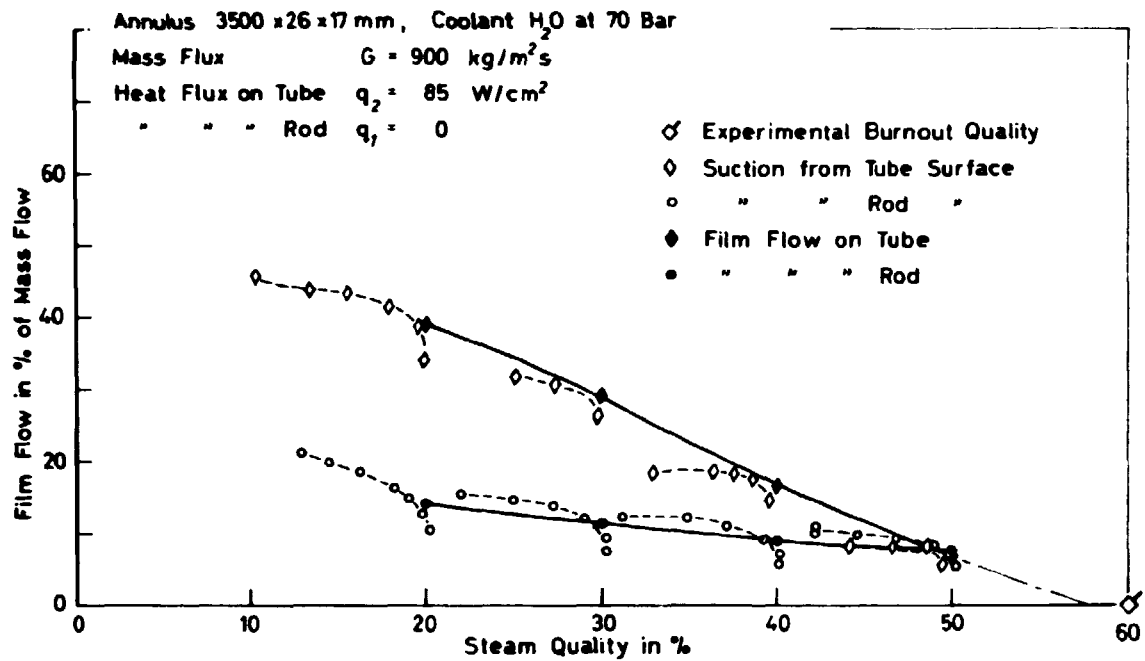


Fig. 5.2.a.

Examples of Film Flow Measurements

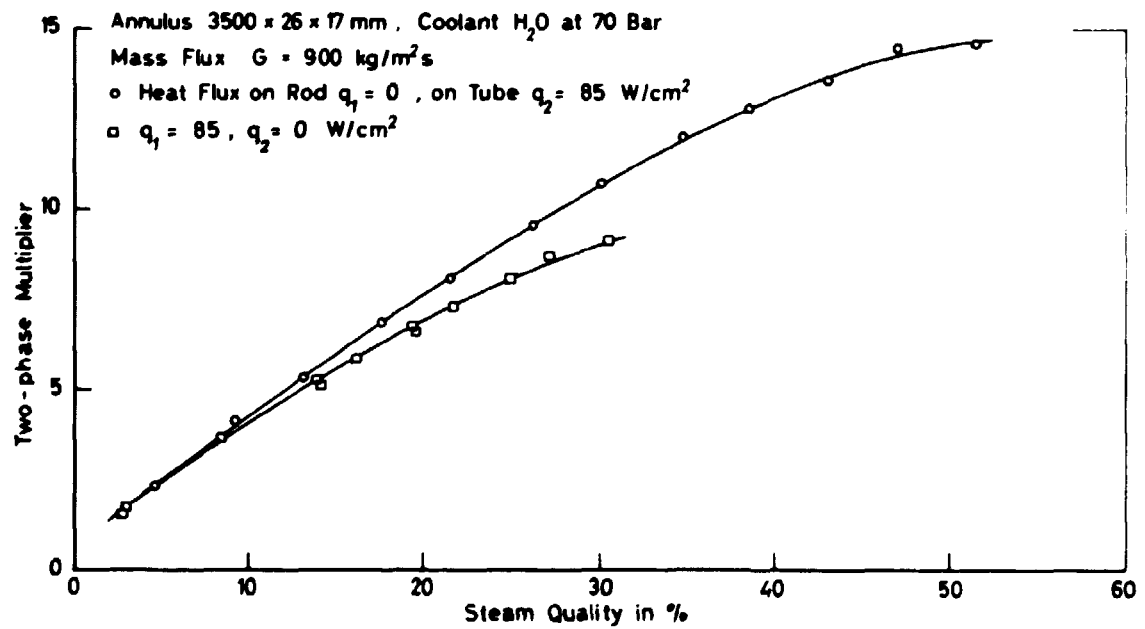


Fig. 5.2.b.

Examples of Pressure Drop Measurements

Briefly the method of film flow measurements is to suck off the film through a perforation in the channel wall at the outlet. The mass flows of water and steam are determined by heat balances, and are plotted versus each other in a diagram shown in fig. 5.2a. The dotted lines are the suction curves for outlet steam qualities 20, 30, 40 and 50%. The fully drawn lines are film flows versus outlet steam quality. Burnout will take place on the tube, and an extrapolation of the tube film flow to zero using the dryout hypothesis predicts a burnout steam quality of 58%. At a corresponding direct measurement of the burnout by a usual bridge type burnout detector the steam quality was found to be 60%, thus confirming the theory.

Examples of pressure drop measurements are shown in fig. 5.2b. The two-phase friction multiplier, which is the ratio between the frictional pressure gradient in the two-phase and the corresponding single-phase case, is plotted versus the steam quality. It is seen, that the heat flux distribution also has a significant effect on the two-phase friction multiplier.

### 5.3 Rewetting of a Hot Rod

A test rig for measuring quench front velocities on an electrically heated rod has been erected. Several test runs have been carried out. The results obtained have successfully been compared with available world data.

This work was a thesis for the Master of Science degree at the Technical University of Denmark.



Fig. 5.3 Quench front.

## 6. SECTION OF DYNAMICS

### 6.0 Introduction

The work in the Section of Dynamics is concentrated on development of dynamic models of nuclear power plants and of the separate components, particularly the reactor. The models are used to study transients both for normal and abnormal working conditions, and the results may be used for safety evaluation of the plant. Further, the models are used for control system investigations.

In the past year the main efforts have been placed on the following subjects:

1. A one-dimensional model of a PWR power plant.
2. Study of rod ejection transients in a BWR.
3. Control system investigations.

### 6.1 A One-dimensional Model of a PWR Power Plant

The basic version of the model running on a hybrid computer and working in real time has been reported in Risø Report no. 318. It is now being used for control system investigations.

The model has further been programmed for the simulation system DYSYS on the Burroughs computer as mentioned in the previous annual report. Thereby it has been possible to make the model more detailed, so it now consists of a one-dimensional reactor model with two primary cooling loops, each with a pump and a steam generator. The two steam generators deliver steam in parallel to one turbine consisting of a high and a low pressure section coupled to feedwater heaters. The steam generators are represented by one-dimensional models.

The DYSYS-model has been tested by calculation of several severe abnormal transients as: Fast power reduction, loss of turbine load, and loss of one and of two primary pumps. Fig. 6.1 gives two examples of transients in the nuclear power for loss of primary pumps; the upper curve for loss of one pump and the lower curve for loss of two pumps.

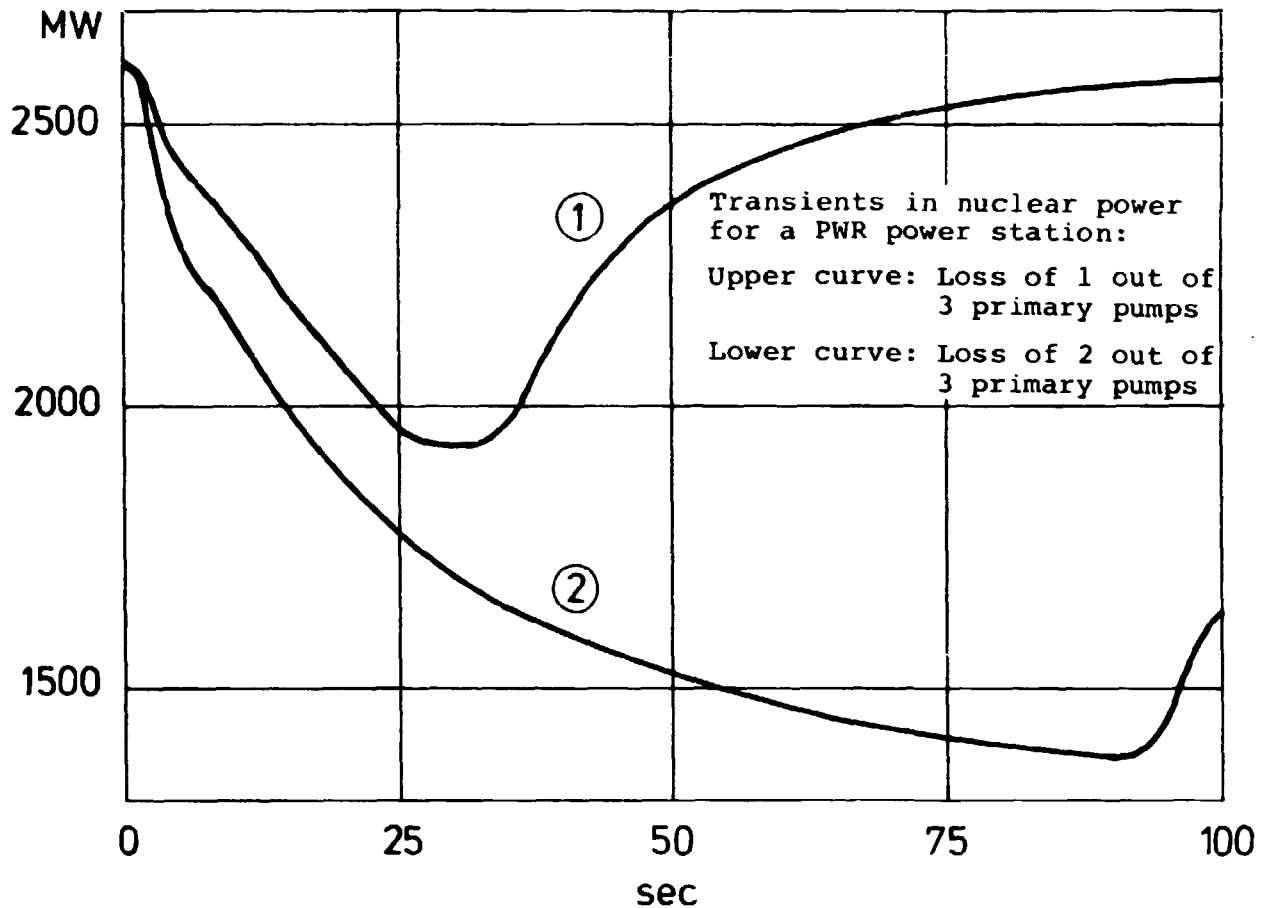


Fig. 6.1.

## 6.2 BWR Control Rod Ejection Accident Analysis

This accident which belongs to the group of rare events in BWR's has been studied by means of a detailed model of the reactor core. The model simulates the conditions in the fuel and moderator when one control rod is ejected from the core due to a postulated break of a control rod drive house from the reactor tank.

The most significant features of the model are:

- a three-dimensional neutronic model of the reactor core based on the nodal theory implying "1½" neutron energy groups and 6 delayed neutron groups
- a one-dimensional fuel and cladding model
- a hydraulic model for the core which has a number of parallel one-dimensional fuel channels coupled in top and bottom and one recirculation loop.
- a number of disturbances: rod movements, changes of pump speed, may be prescribed.

In this analysis the disturbance is a control rod movement. The data for this is derived with a special model for the control rod structure being accelerated outwards by the reactor pressure.

An example with a control rod ejection from a reactor containing 400 fuel elements, each with 64 fuel pins, is given below. Nominal power is 1.3 GW. At the beginning of the transient the reactor conditions correspond to hot critical (power equals  $10^{-6}$  of nominal, 30% pump speed, and pressure 70 bars). The ejected rod is situated centrally in the core and fully inserted with a reactivity of approximately 2%. During the transient it is assumed that the internal structures are unchanged, a perhaps questionable assumption.

The results are given in fig. 6.2.1 - 6.2.4.

Fig. 6.2.1 shows the total fission power and various components in which it may be separated. The prompt power is the part which is released in the fuel. A part of the radiation from the fissions will, however, be transferred directly to the coolant and converted to heat there. The component of the power which is released in the fuel will reach the coolant by conduction delayed by the heat capacity of the fuel, and the power in the coolant is thus made of two parts namely the convective and the direct (radiative) power.

Fig. 6.2.2 shows the flux peaking which is rather high at the time of maximum reactivity of the rod. Fig. 6.2.3 shows how the liquid is expelled from the hot fuel channels both upwards and downwards due to the violent vapour production. This implies that the critical heat flux certainly is exceeded in part of the transient. Fig. 6.2.4 shows, however, that the rate with which the reactor pressure increases, is rather slow and it may be assumed that no damage will be exerted to the reactor tank if the safety valves do function correctly, while no estimate concerning the damage to the internal structures has been made.

The fuel temperatures of this transient will exceed  $1100^{\circ}\text{C}$  in the central fuel boxes and  $1050^{\circ}\text{C}$  in the neighbours. Due to the high heat fluxes at voids near 1 the cladding temperatures will exceed the sputtering temperature before rewetting can take place and therefore it is estimated that a great number

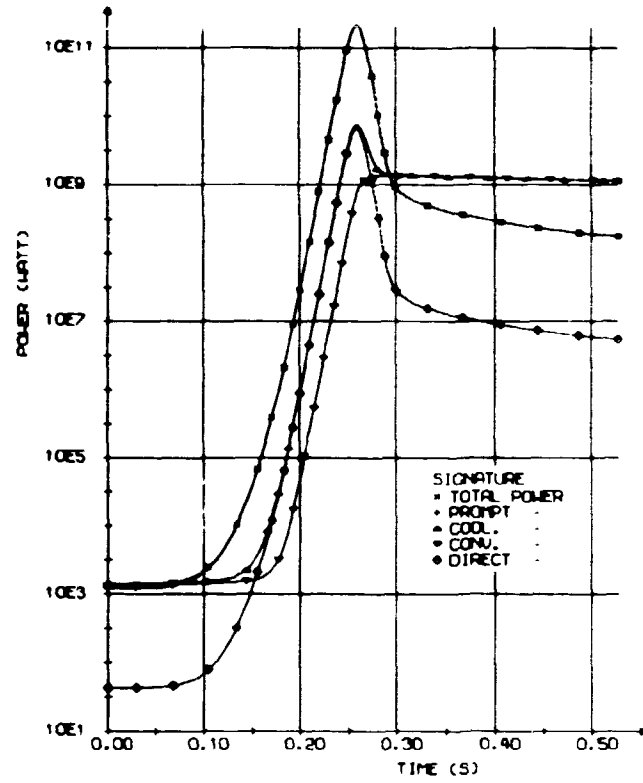


Fig. 6.2.1. Rod ejection transient power generation.

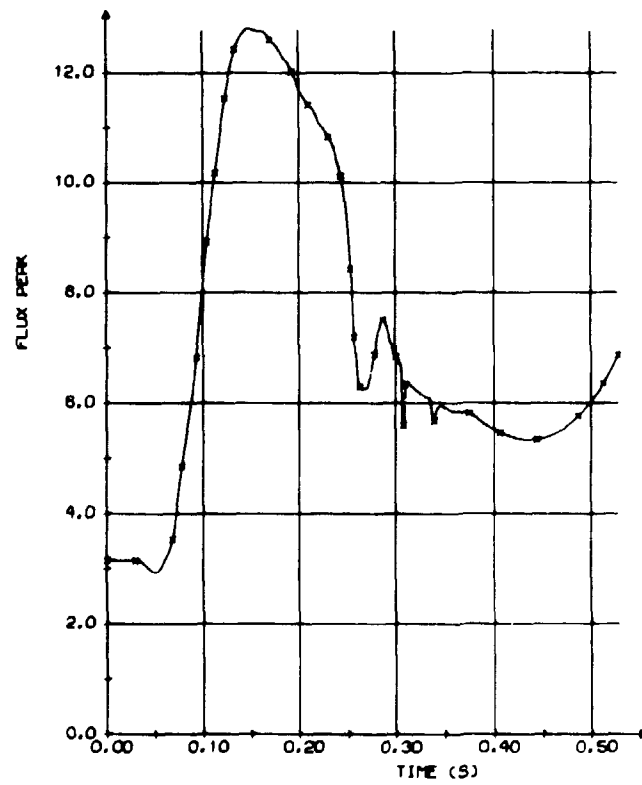


Fig. 6.2.2. Rod ejection peak to average flux ratio.

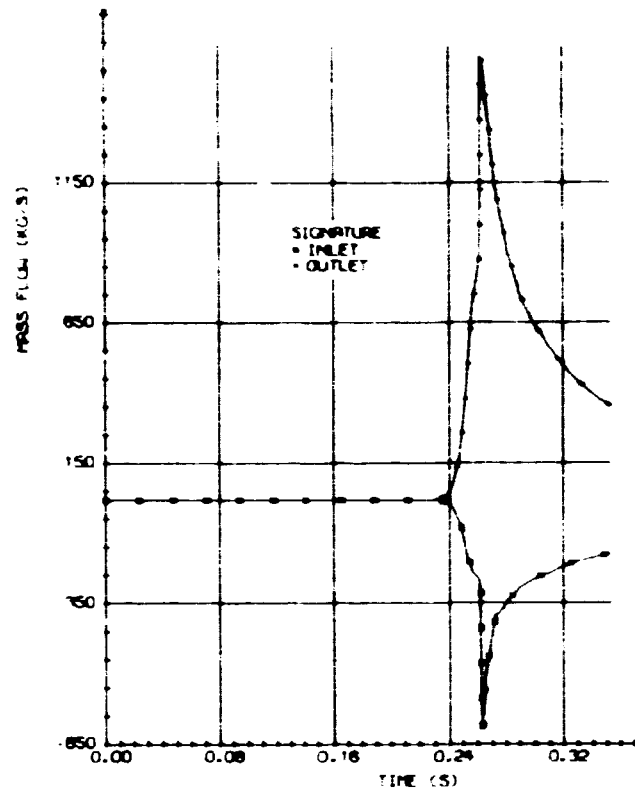


Fig. 6.2.3. Hot channel coolant inlet and outlet mass flows.

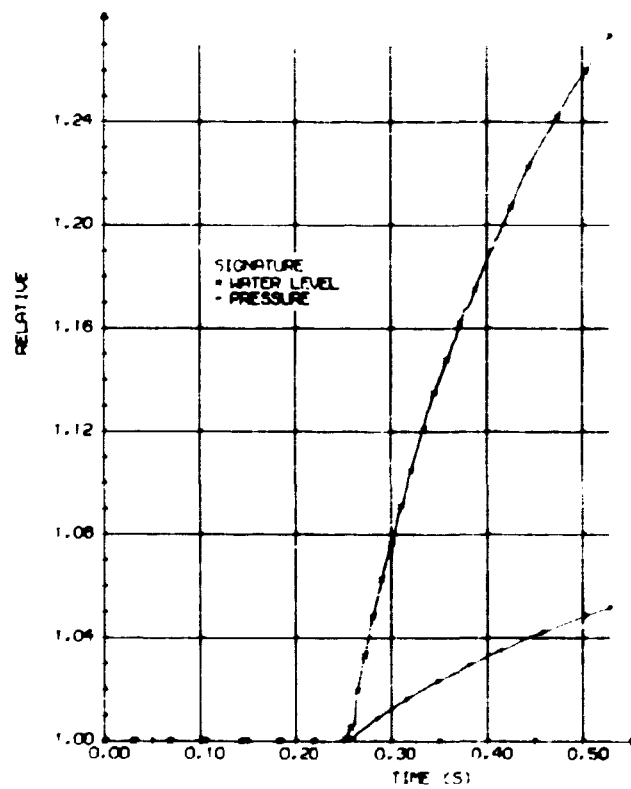


Fig. 6.2.4. Water level and reactor pressure relative to initial value.



of fuel pins (hundreds) will perforate and release radioactivity into the cooling water.

In these calculations several assumptions are involved, the validity of which is uncertain. A sensitivity study was for that reason made, where the dependency on the control rod worth, the rod velocity, and the initial power level, was studied. The rod parameters had no great influence on the transient, whereas the initial power level was important in that a higher initial power level made the transient less violent.

Details of this study may be found in Risø Report no. 344.

### 6.3 Control Theory

The work on Self-organizing control systems has been continued. The theoretical studies based on automata theory which has been reported earlier is now supplemented with a more practical oriented analysis of a specific system.

The self-organizing control system (SOC) analyzed is developed by R.L. Barron, USA. The SOC employs random search to control the plant and modify continuously its functional characteristics according to an overall goal. The controller is non-linear and an accurate analysis of its dynamical properties cannot be accomplished.

The analysis made of Barron's SOC describe the static properties of the controller and covers both single and multi-variable applications. Due to the special functions of the SOC its advantages are only recognized in the multivariable case, but in the analysis the results of the single variable case is the basis for the multivariable analyses. The analysis provides a model of the SOC which relate statistical averages of the internal variables in the controller.

The following results have been obtained in this study

- a) Tuning rules have been developed for the SOC. These rules apply equally for the singlevariable and the multivariable case.
- b) Redundant control systems using SOC's provide more functional flexibility than control systems based on conventional techniques. This result is derived from the analysis of the multivariable case, and has been verified by experiments on an analog computer.

- c) When used as a decoupling controller the SOC is able to compensate for certain changes of the plant gain matrix. This result indicate that the SOC can be used with advantage for the control of nonlinear multivariable plants.

Another result of the analysis is the formulation of the basic constraints on SOC capability. The nature of these constraints indicate that further development of the SOC is necessary. The analysis made provide a theoretical basis for such a work.

## 7. THE DANISH REACTOR NO 1

### 7.0. Introduction

The reactor has mainly been used for neutron radiography and training purposes.

The reactor was used for teaching in nine courses lasting 2-20 days for students from the Danish Technical University, the University of Copenhagen, the Technical University in Lund, Sweden, and the Danish Royal Veterinary and Agriculture University.

### 7.1. Neutron Radiography

Several fuel pins were radiographed by use of Dysprosium foils. The use of two lead containers facilitated the transportation of the fuel pins between Hot Cell and DR 1.

Non-radioactive objects are radiographed by the direct method, where the film is situated directly in the neutron beam. In this way the resolution is improved, especially when a Gadolinium foil is used as converter.

### 7.2. Germanium Detector

The activities of several cobalt and nickel wires were measured for the Isotope- and Metallurgy department. Further the thermal and fast neutron fluence were calculated.

Two 10 cm thick lead cells were moved from the Hot Cell to the DR 1 hall. One of the cells is used to decontaminate the wires, and the other to measure the activities by means of an ionisation chamber.

### 7.3. Pile Oscillator

The equipment was tested by measuring the signals for different isotopes. Both the local and the global oscillators give results which are in good agreement with expected values.

#### 7.4. Mössbauer Effect

The spectrometer was used to determine the heating-up temperatures of some soil samples from Glozel, France. Temperatures were determined by the changes of peak positions and line broadenings.

## 8. ECONOMIC STUDIES FOR POWER PLANTS

In view of the many statements made on the economics of nuclear power plants compared to conventional fossil fired power plants, a study on the economics of various types of power plants, built and operated under Danish conditions, has been initiated. The main emphasis of the study is put on investment and financing aspects of the decision on which type of base load electricity generating plant to build for entering service in 1987; this is the earliest point in time that a Danish nuclear power plant can be connected to the grid. The economic aspects of the enlargement of the Danish power production system are shortly redressed. The economics of plants for electricity production only versus of plants for combined electricity and district heating production will be dealt with in a study just started. Selected results from the present study are given below. The full study is contained in a Risø Report to be published shortly.

The assumptions applied in the study rest on a rather extensive search for available data, combined with a hopefully intelligent guess on the future developments. The huge number of assumptions will not be stated here, it shall only be mentioned that it is assumed that:

1. A nuclear power plant will be built on a new site, while a conventional plant will be built on an existing site, where plants of the same type are already operating.
2. The price of uranium will increase in real terms, while the price of fossil fuels will stay constant in real terms.
3. A nuclear power plant will have a more pronounced running-in period than a conventional fossil fired plant.

The significance of uncertainties in data is studied in a sensitivity analysis.

Fig. 8.1 shows for the four types of power plants investigated, the lifetime average kWh-costs and its split-up on main com-

ponents. Fig. 8.2 shows a split-up of the fuel cycle costs.

Equal lifetime average kWh-costs for a 900 MW LWR power plant with reprocessing of the spent fuel and a 600 MW coal/oil fired power plant without SO<sub>2</sub> control will be obtained at:

1. A 90% increase in construction costs for LWR plants; all other data unchanged.
2. A 200% increase in construction costs for LWR plants as well as for coal/oil fired plants; all other data unchanged.
3. An increase in the price of uranium by 7% p.a. in real terms from 38 \$/lb at mid 1976 (~80 1976-\$/lb in 1987 and ~190 1976-\$/lb in 2000); all other data unchanged.
4. An increase in the price of enrichment service by 6% p.a. in real terms from 100 \$/SWU in 1976 (~190 1976-\$/SWU in 1987 and ~ 400 1976-\$/SWU in 2000); all other data unchanged.

Fig. 8.3 shows lifetime average kWh-costs as function of load factor for 2 different values of the forced outage rate. The curves are evaluated for a constant electricity production and for a fixed supply reliability. The lower the load factors, the greater the amount of electricity to be produced on marginal power plants in the production system will have to be; the higher the forced outage rate, the greater the installed capacity will have to be. A high forced outage rate is compensated by installation of further capacity in plants characterized by the same data. According to fig. 8.3 the lifetime average kWh-costs for a 900 MW LWR plant and for a 600 MW conventional coal/oil fired plant without SO<sub>2</sub> control, both brought into service in 1987, will be equal at load factors for the LWR plant of 65% of the reference values for this plant type, all other data, incl. load factors for the coal/oil fired plant, being as in the reference case. It is further noticed, that for LWR- and CANDU plants and for coal/oil fired plants without SO<sub>2</sub> control the lifetime average kWh-costs at high load factors are lower for a high forced outage rate than for a low forced outage rate, when the load factors are the same. The reason being, that at high load factors, the total costs per kWh for a new plant of the said types will be lower than the variable costs for mar-

ginal plants.

Fig. 8.4 shows the kWh-price as function of time in inflation-ary money, when it is assumed that:

1. The yearly inflation is 8%.
2. The construction costs will be financed by a loan carrying a yearly interest rate of 12.3%; the loan is paid back in equal yearly installments over the first 15 years of plant operation.
3. The running expences are covered by the yearly receipts.
4. The costs of dismantling the power plants after end of service are covered by savings. The money are collected from the consumers in equal yearly amounts over the last 15 years of plant operation, and invested, the interest rate being 12.3% p.a.

Key:

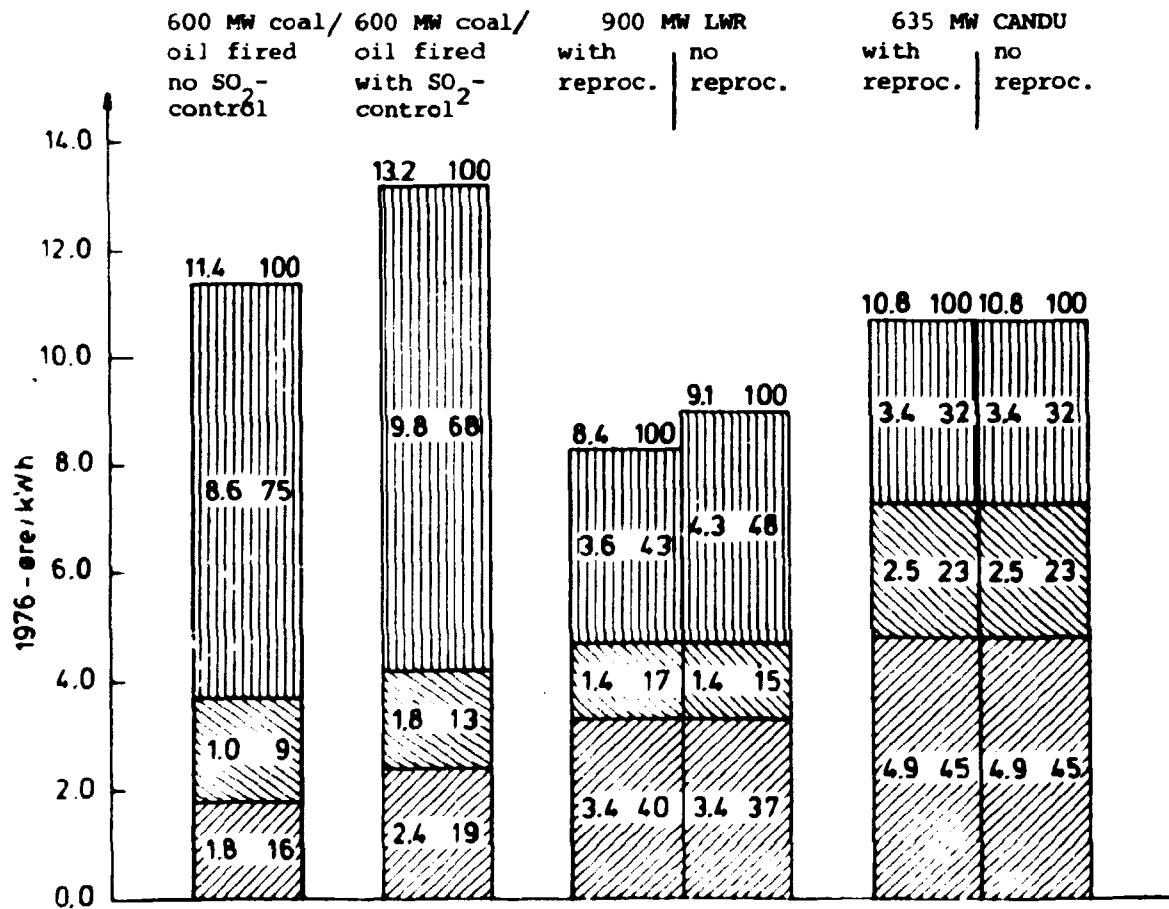
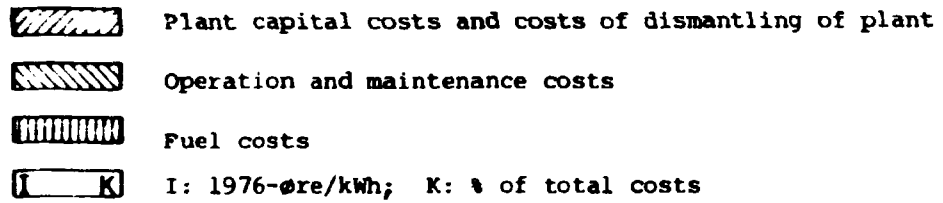


Fig. 8.1. Breakdown of Projected Levelized Power Costs



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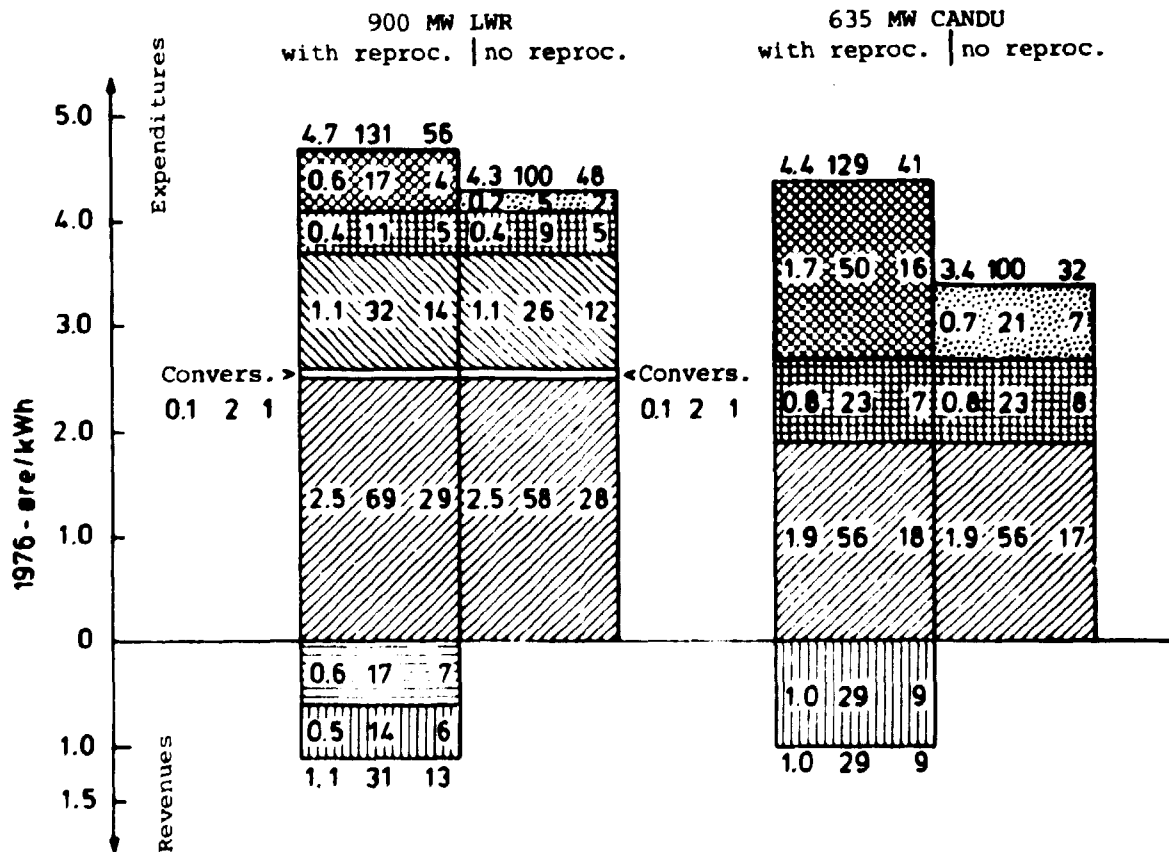
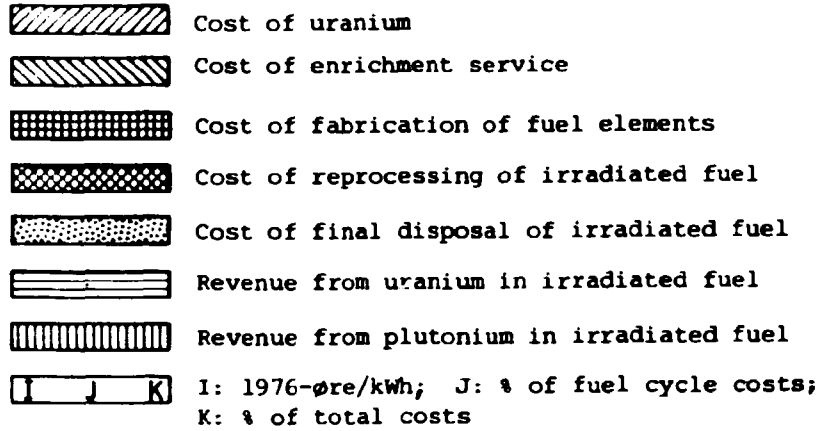


Fig. 8.2. Breakdown of Projected Levelized Fuel Cycle Costs

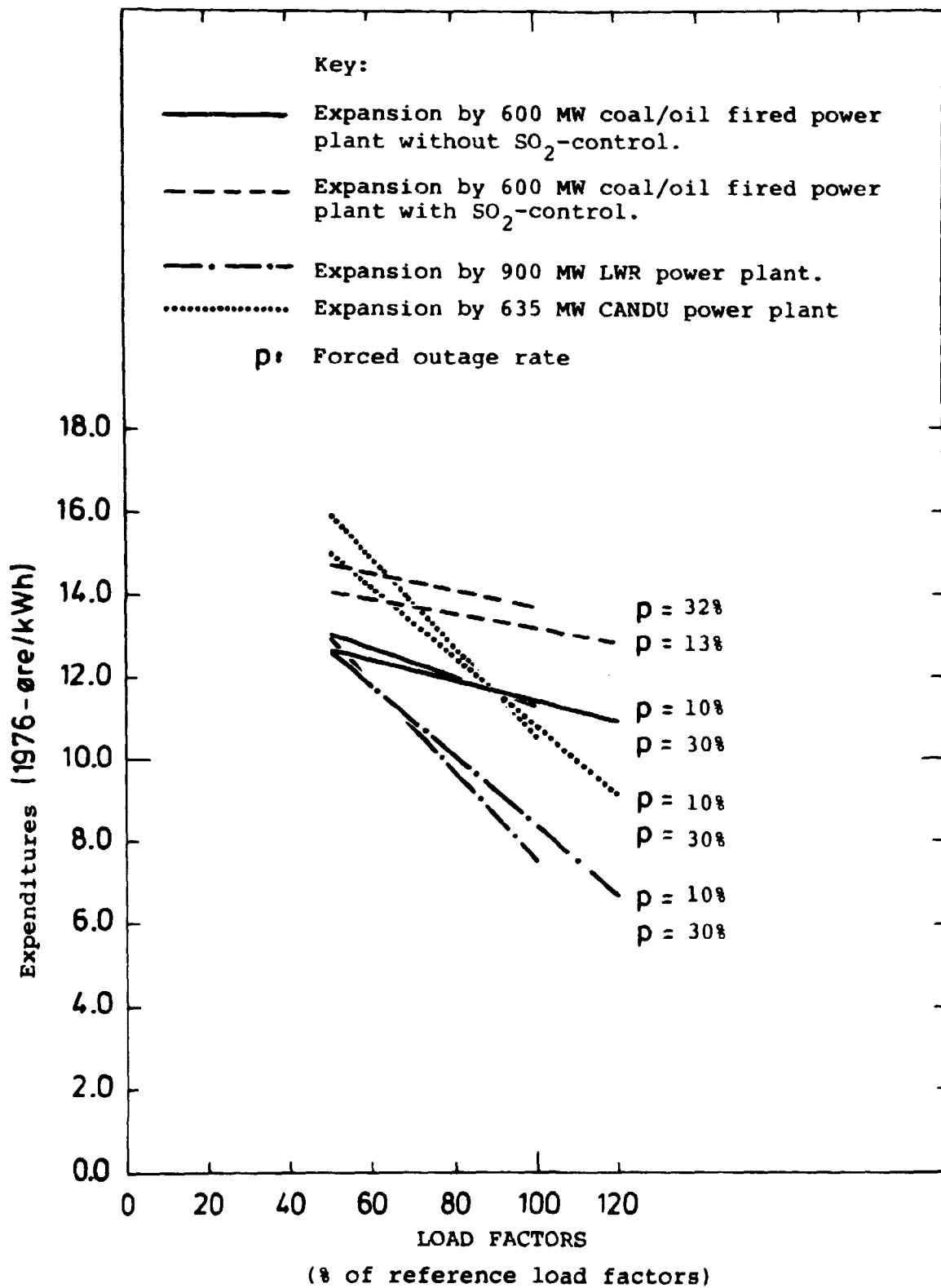


Fig. 8.3. Projected Levelized Power Costs in Dependence of the Load Factor for 2 Values of the Forced Outage Rate.

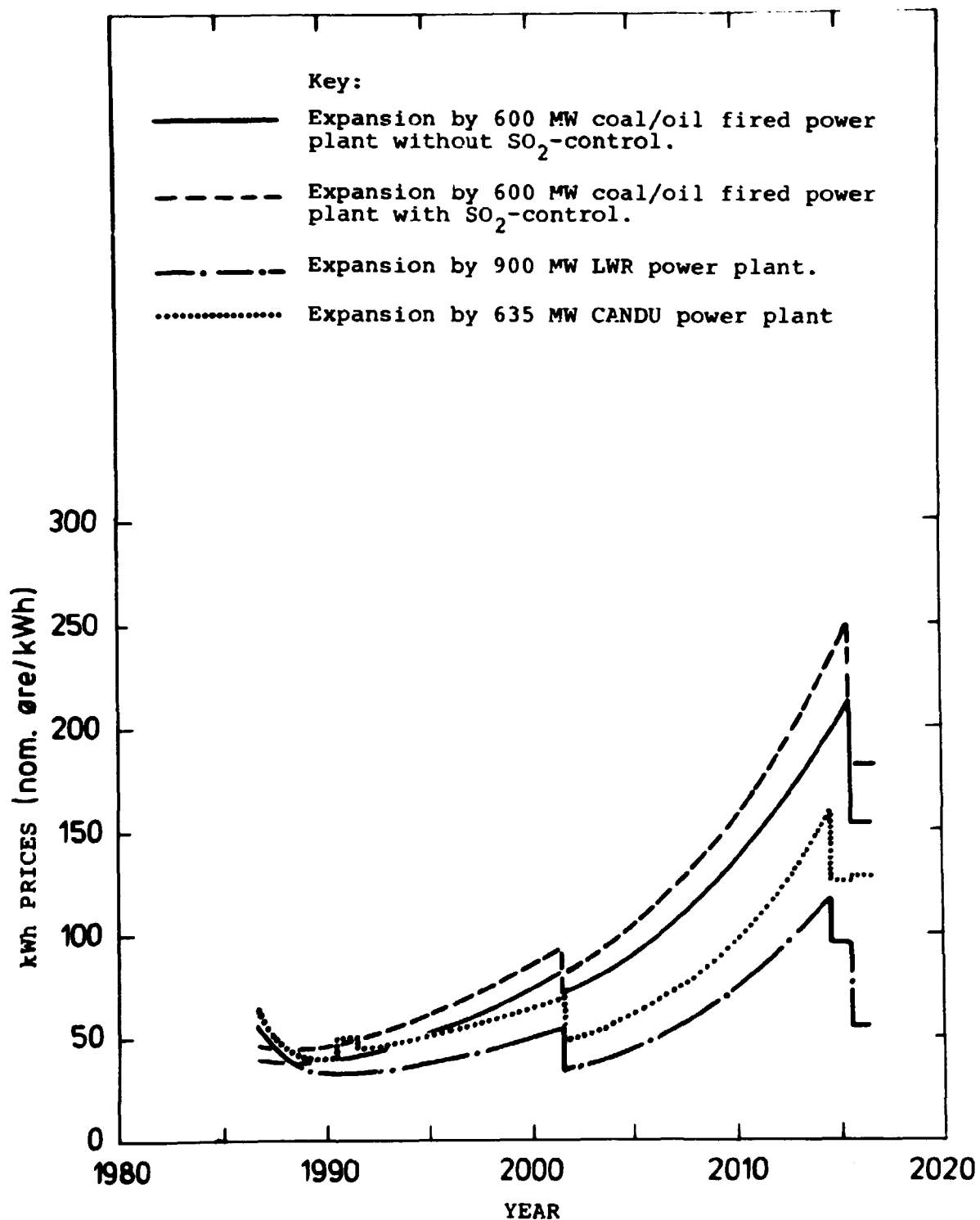


Fig. 8.4. kWh Prices for a Given Set of Financing Conditions (see text).

## 9. PUBLICATIONS

### 1. Risø Reports

Risø Report No. 337 O. Rathmann. Crystalline Electrical Fields in Dilute Rare Earths Studied by Neutron Scattering. March 1976.

Risø Report No. 315. Morten Lind, Investigation of a Class of Self-Organizing Control Systems, June 1976.

### 2. Various Works Published

Iqbal Ahmed and Vagn Mejdahl, Determination of Pottery Firing Temperatures by Mössbauer Spectroscopy, March 1976.

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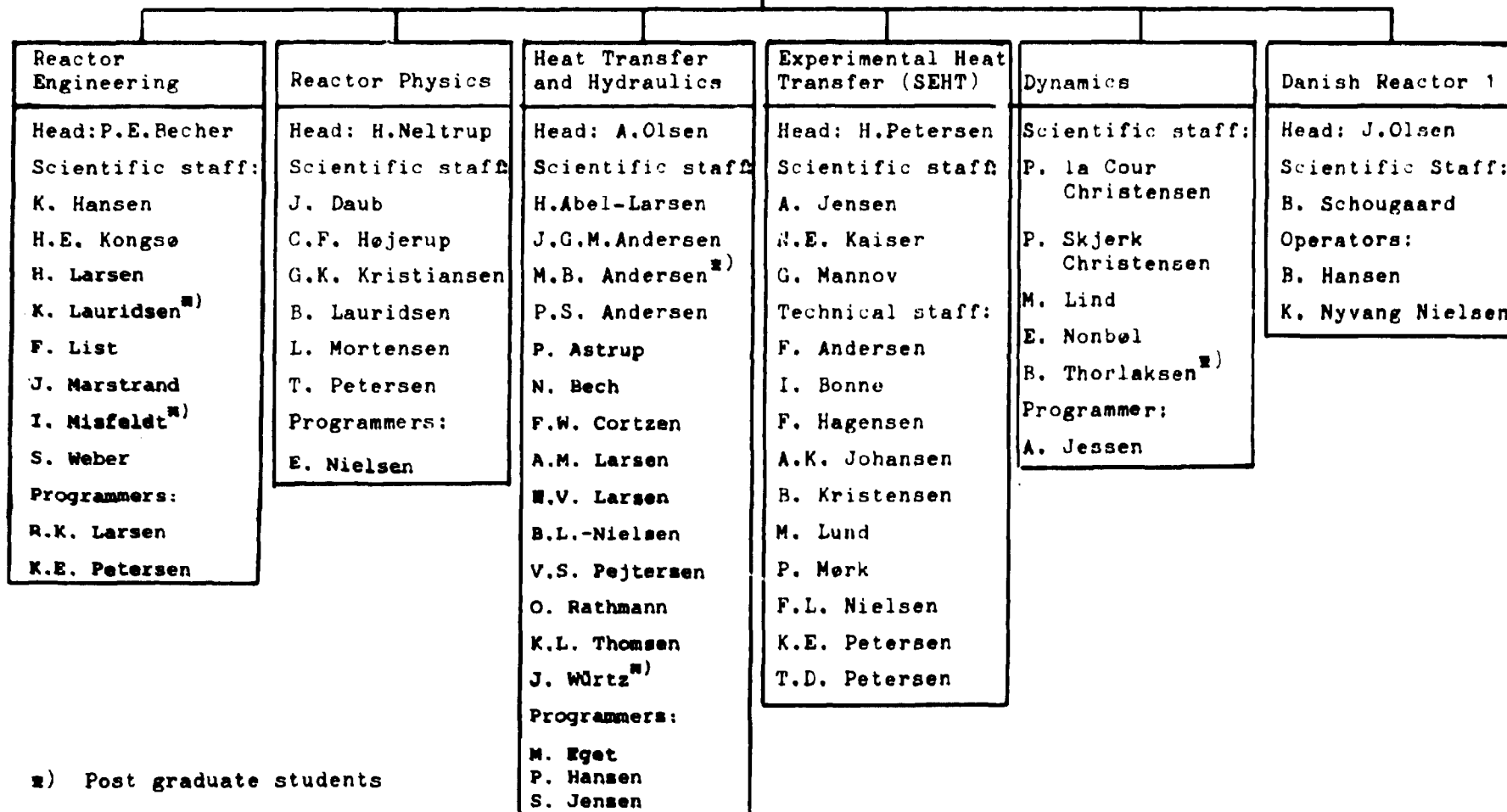
1. Februar 1977

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